Boiling Water Reactor
GE BWR/4 Technology
Technology Manual

Chapter 12.3

Refueling and Vessel Servicing System

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12.3 REFUELING AND VESSEL SERVICING SYSTEM

12.3.1 Introduction

The purpose of the Refueling and Vessel Servicing System is to provide facilities for the handling and storage of new and spent fuel and to provide equipment for vessel refueling and servicing of vessel internal components. The functional classification of the Refueling and Vessel Servicing System is that of a power generation system.

12.3.2 System Description

The Refueling and Vessel Servicing System consists of three subsystems: the Fuel Handling and Storage Subsystem, the Fuel Servicing Subsystem, and the Vessel Servicing Subsystem.

The Fuel Handling and Storage Subsystem provides the equipment necessary to receive, prepare, and install new fuel as well as the equipment for removing, preparing, and shipping spent fuel. This subsystem provides storage facilities for new fuel, spent fuel, and other irradiated vessel components.

The Fuel-Servicing Subsystem provides the equipment necessary to remove, store, inspect, and install various internal and external components of the reactor vessel. The reactor vessel is serviced from both the containment refueling floor and from the under vessel service area.

12.3.3 Component Description

The major components of the Refueling and Vessel Servicing System are discussed in the paragraphs that follow.

12.3.3.1 Fuel Handling and Storage Subsystem

The new fuel arrives at the site packed in wooden crates. Each crate contains two fuel bundles, both enclosed in a steel container which supports the entire length of the bundles. The new fuel is raised from the receiving area up to the fuel handling area by the new fuel bridge crane (Figure 12.3-1).

Once new fuel has been received, it is placed in the new fuel storage vault. Located within this vault are 24 storage racks each containing up to 10 fuel assemblies. This storage capacity represents about 30% of the reactor core load.

The arrangement of the fuel assemblies in the new fuel storage vault ensures that Keff will not exceed 0.95 in both a dry vault condition or when the vault is completely flooded with water. In the dry condition, Keff is maintained <0.95 because of under moderation. In the flooded condition, the geometry of the fuel storage array ensures that Keff will remain <0.95; thus, subcriticality does not depend upon the presence of any neutron absorbing materials.

The new fuel is now inspected and channeled as indicated in the description of the Fuel Servicing Subsystem. When the new fuel is to be used in the reactor, the new fuel bridge crane removes the fuel bundles from the new fuel storage vault and places them in the fuel preparation machine where they are fitted with channels. The channeled new fuel is then placed in the spent fuel pool storage racks by the spent fuel gantry crane.

The spent fuel gantry crane, shown in Figure 12.3-2, is the primary means of transporting fuel in the fuel handling area. Its range includes the spent fuel pool, cask storage pool, and transfer pool. The spent fuel gantry crane supports a refueling grapple and two auxiliary hoists, the frame mounted hoist, and the monorail hoist. The fuel grapple is suspended from a trolley system that can traverse the width of the platform and is mounted on the upper structure of the gantry.

The fuel grapple consists of a telescopic, stainless steel mast, an air operated grapple at the end of the mast, and the main hoist which raises or lowers the grapple. The mast consists of a fixed section and three telescopic sections which allow operation of the grapple between 8 feet and 56 feet below the top of the platform tracks. The sections have a triangular cross section. The 8 foot upper travel limit ensures adequate shielding between the fuel on the grapple and the spent fuel gantry.

Suspended from the trolley is the operator's cab which contains all the controls necessary to control the operations of the gantry, the fuel grapple, the frame mounted hoist, and the trolley.

The spent fuel storage racks provide a place in the fuel storage pool for storing spent fuel removed from the reactor vessel. These are top entry racks designed to maintain the spent fuel in a geometry that precludes the possibility of criticality under normal and abnormal conditions. The spacers and upper tie plate of the fuel assemblies rest against the rack to provide lateral support. Each standard spent fuel rack stores 10 fuel assemblies.

The spent fuel pool is designed so that no single failure of structures or equipment will cause inability to maintain irradiated fuel submerged in water, or to reestablished normal fuel pool water level. In order to limit the possibility of pool leakage, the pool is lined with stainless steel. In addition to providing a high degree of integrity, the lining is designed to withstand abuse that might occur when equipment is moved about. No inlets, outlets, or drains are provided that might permit the pool to be drained below a safe shielding level. Lines extending below this level are equipped with siphon breakers, check valves, or other suitable devices to prevent inadvertent pool drainage.

The Fuel Pool Cooling and Cleanup (FPCC) System, discussed in Section 12.1, removes decay heat from the fuel assemblies and maintains clarity and purity of the pool water. Fuel, control rods, and other small items are transferred between the refueling floor and the fuel handling area using the Fuel Transfer (FT) System, discussed in Section 12.2. Once the new fuel has been transferred to the upper containment pool, it is removed from the transfer mechanism by the refueling platform and is either placed directly in the vessel or into the temporary fuel storage racks in the upper containment pool (Figure 12.3-3).

The refueling platform is used as the principal means of transporting fuel assemblies between the reactor well and the containment fuel storage pool. It also serves as a hoist and transport device. It provides an operator with a work surface for almost all the other servicing operations. The platform travels on track extending along each side of the reactor well and pool and supports the refueling grapple and auxiliary hoists. The platform design permits travel over the safety railings placed around the pools. The platform supports a refueling grapple and two auxiliary hoists, the frame mounted hoist, and the monorail hoist. The fuel grapple is suspended from a trolley system that can traverse the width of the platform. The trolley system is mounted on the upper structure of the platform. The fuel grapple is identical to the fuel grapple of the spent fuel gantry crane.

To move fuel, the fuel grapple is aligned over the fuel assembly, lowered, and attached to the fuel bundle bail. The fuel bundle is then raised out of the core, moved through the refueling slot in the reactor well wall to the containment fuel storage pool, positioned over the storage rack, and lowered into the rack. The new fuel is then moved from the containment storage pool to the reactor vessel in the same manner.

The upper containment, fuel storage pool is equipped with spent fuel racks that provide a temporary storage capacity that represents 25% of one core fuel load. This storage space is used during refueling operations only, and no fuel is stored there during plant operation.

Spent fuel removed from the reactor is transferred off site in a special shipping cask, shown in Figure 12.3-4. The empty cask arrives at the receiving area on a specially designed shipping flatbed (truck or railroad car). Domestic water is used to wash the road dirt from the cask, and if the inspection shows that the radioactivity level of the cask exceeds 10 CFR 20 limits, the cask is washed again with demineralized water.

The spent fuel cask crane upends the cask, raises it to the fuel handling area, and transfers it to the cask storage pool.

The spent fuel cask crane cannot travel over the spent fuel storage pool, a feature which precludes a cask dropping onto irradiated fuel. As the cask is lowered into the cask storage pool, the water is drained so that the water level is maintained. After the cask is set on the cask storage pool floor, the cask storage pool is refilled and the two pool separation gates to the fuel storage pool are opened, using the new fuel bridge crane where necessary.

Once the cask is loaded with the spent fuel gantry crane, the gates are closed and the water level is lowered to the top of the cask. As the cask is raised, the water level is maintained just below the top of the cask, until the pool is filled and the cask is out of the water.

Once the cask has cleared the pool, it is transferred to the cask washdown area where its internals are flushed and its exterior is scrubbed down with demineralized water. If the radioactivity emitted from the cask is in compliance with 10 CFR 20 limits, the cask is transferred to the flatbed. When the cask cooling system is connected to the cask, the cask is ready for shipment to a fuel processing plant.

12.3.3.2 Fuel Servicing Subsystem

The Fuel Servicing Subsystem includes the equipment necessary for the inspection, channeling, and sampling of the fuel assemblies.

The actual inspection of the new fuel is normally deferred until all the reusable containers are emptied and the area around the new fuel storage vault is cleared. At that time the individual fuel bundles are removed from the vault and are inserted in the new fuel inspection stand to be inspected.

The new fuel inspection stand, shown in Figure 12.3-5, consists of an upper and a lower platform, each with guard rails. In use, the inspection stand accommodates two inspectors, one on each platform. The stand holds two fuel bundles mounted on rotatable bearing surfaces. This arrangement permits the inspector on the upper platform to inspect the top portion of each of the two fuel bundles and the inspector on the lower platform to inspect both lower halves. The rotating bearing surfaces are equipped with hand operated, locking devices to hold the bundles in the desired quadrant positions during inspection. The clamp plates on the top of the stand hold the bundles in place.

Fuel bundle inspection consists of first a visual check of the bundle to see whether fingermarks, grease, dust, etc. are present. Any such marks are removed using commercial grade acetone and lint free rags. Also, the visual inspection ensures that there are no dents, cracks, or other types of physical damage in evidence on the handle, base, etc. as a result of rough handling during shipment. Next, all plastic shipping spacers are removed from between the fuel bundle rods so that no plastic spacer particles remain. Inaccessible chips of foam plastic are removed with oil free, filtered, compressed air.

Other checks include fuel rod seating, spacer checks, tie rod nut locking check, and rod clearance checks.

Preparation of the fuel bundles for loading in the reactor core requires the installation of channels on the new fuel bundles. Installation of new (or irradiated but reusable) channels is performed with the aid of the fuel preparation machines located in the fuel storage pool (Figure 12.3-6).

The fuel preparation machine is used both for stripping reusable channels from the spent fuel bundles and for rechanneling the new fuel. The machine is also used with a fuel inspection fixture to provide an underwater inspection capability. Two fuel preparation machines are mounted on a wall of the spent fuel pool.

A channel storage rack is provided to hold up to 20 fuel channels to be used during fuel bundle preparation. The channel storage rack is located in the spent fuel pool and centered between the two fuel preparation machines.

The channel handling tool is suspended from a channel handling boom located on the side of the spent fuel pool. The channel is oriented so that the corner with the drilled tab is over the threaded boss on the upper fuel bundle tie plate, then carefully lowered over the fuel bundle.

The channel is secured to the fuel bundle by means of a channel fastener (spring clip) threaded through the channel tab and into the fuel bundle boss. A channel bolt wrench is used for removing and installing the channel fastener while the fuel bundle is held in the fuel preparation machine.

Jib cranes, consisting of a motor driven swing boom monorail and a motor driven trolley with a 100 pound capacity electric hoist, are mounted along the edge of the spent fuel storage pool and the upper containment pool to be used during refueling operations. Use of the jib crane leaves the refueling platform or spent fuel gantry crane free to perform general fuel shuffling operations and still permit uninterrupted fuel preparation in the work area. A jib crane is shown in Figure 12.3-7.

The jib cranes are capable of moving fuel or other components when fitted with the general purpose grapple. This grapple is designed to carry fuel elements, control rods, and other objects having a bail handle similar to that of a fuel assembly and weighing no more than 1000 pounds.

During reactor operation, the offgas radiation level is monitored. If a rise in offgas activity has been noted, the reactor core may be sampled during shutdown to locate any leaking fuel assemblies. The fuel sampler or sipper shown in Figure 12.3-8, isolates up to a 16 bundle array in the core. This stops water circulation through the bundles and allows fission products to concentrate if a bundle is defective. After 10 minutes, a water sample is taken for fission product analysis. If a defective bundle is found, it is transferred to the fuel storage pool and stored in a special defective fuel storage container to minimize background activity in the storage pool.

The sipping head adapter for the defective fuel storage container is used to sample an individual fuel bundle for leakage. The sipping head fits tightly over the container, and necessary fittings and hoses are provided to adapt the fuel bundle sampler pump for use with the sipping head. The defective fuel storage containers are placed in one of the control rod and defective fuel storage racks.

Each control rod and defective fuel storage rack consists of two rows of four cylindrical containers plus a single cylinder centered at each end of the rack. In addition to control rods and defective fuel, control rod guide tubes can be stored in these racks. Holes in the bottom of each storage rack allow free circulation of water through each cylinder.

12.3.3.3 Vessel Servicing Subsystem

The Vessel Servicing Subsystem provides the equipment necessary to remove, store, inspect, and install various internal and external components of the reactor vessel. The reactor vessel is serviced from both the containment refueling floor and from the under vessel service area. The upper containment pool is a rectangular stainless steel lined pool crossing the top of the drywell. The water in this pool provides shielding in the refueling floor during normal reactor operation. The upper containment pool is divided into five areas: the steam separator storage area, the reactor well, the steam dryer storage area, the fuel storage area, and the fuel transfer pool.

Laydown areas for the drywell head, vessel head, and vessel head insulation are designated on the containment refueling floor on either side of the upper pool.

The polar crane is a bridge type crane spanning the interior of the containment. It rides on a circular rail around the containment wall, and its services include handling the drywell head, vessel head, dryer, and separator during refueling. The polar crane has a 125 ton capacity with a 35 ton auxiliary hoist.

In order to perform the required reactor servicing, the reactor is shut down according to prescribed procedures, and during cooldown the reactor pressure vessel is vented and filled to above the flange level to promote cooling. The reactor well is drained during this time in preparation for drywell and vessel head removal.

Immediately after vessel cooldown and reactor well draining, the work to unbolt the drywell head can begin. The unbolted drywell head is lifted by the polar crane, as shown in Figure 12.3-9, to its appointed storage space on the refueling floor. The drywell seal surface protector is installed before any other activity proceeds in the reactor well area.

After all interfering piping, instrumentation, and necessary vessel head insulation have been removed, the combination vessel head strongback and carousel stud tensioner, shown in Figure 12.3-10, is then transported by the polar crane and positioned on the reactor vessel head. The strongback attaches to the vessel head at four locations by bolts and nuts through lifting eyes on the vessel head and the sling. The four connecting points have a leveling adjustment. The head strongback also incorporates a carousel frame from which the stud tensioners are suspended. Figure 12.3-11 shows the reactor vessel head removal sequence.

The head stud tensioner is used to remove and reassemble the reactor vessel head enclosure retaining nuts. The assembly includes eight stud tensioners suspended from the head strongback and carousel structure by a circular, rail mounted, 1 ton, power hoist for each tensioner. This permits working on eight studs at each operation. The stud tensioner positions over the enclosure stud resting upon the vessel head flange. The tensioner hydraulically places the stud under tension, stretching it to a predetermined amount. The enclosure nut is loosened with a hand operated lever integral with the tensioner. Upon release of the tension, the enclosure nut is hand rotated free from the stud. Tightening the enclosure nut generally involves the reverse procedure. The six studs in line with the fuel transfer gate through the reactor well wall are removed from the vessel and placed in the rack provided for them. The loaded rack is transported to the refueling floor for storage. Removal of these studs provides a path for fuel movement. Access to the fuel assemblies requires the removal of the steam dryer and separator. The steam dryer is removed using the dryer separator strongback, as shown in Figure 12.3-12, and stored in the dryer pool. The dryer is not expected to be highly radioactive and is, therefore, transported in air.

Before removing the steam separator, the steam line plugs, shown in Figure 12.3-13, are installed in the four main steam nozzles from inside the vessel. This is accomplished from the refueling platform using a steam line plug installation tool suspended from the polar crane. The plugs are guided into place and inflated. Thus, the servicing of the safety/relief valves can be accomplished without adding to the critical path refueling time. The shroud head and steam separator assembly can now be removed after the shroud head bolts have been loosened by using the shroud head bolt wrench. The shroud head and separator assembly is removed, as shown in Figure 12.3-14, using the dryer and separator strongback, and is stored in the separator pool. Since the shroud head is expected to be highly radioactive, the separator assembly is transferred so that it remains under water. Once access to the fuel is possible; the refueling platform, upon which is mounted the fuel grapple, is moved into place.

To move fuel assemblies, as shown in Figure 12.3-15, the fuel grapple is aligned over the assembly, lowered, and attached to the fuel assembly bail handle. The assembly is raised out of the reactor vessel and moved through the reactor well gate slot to the fuel storage area, where it is either positioned over the storage racks and lowered into the rack for temporary holding or placed in the transfer tube for transfer to the fuel handling area.

A control rod is removed from its cell by first removing two diametrically opposite fuel assemblies and then inserting a blade guide, shown in Figure 12.3-16, in their place. This is followed by the removal of the remaining two fuel assemblies, the lowering of the control rod to its withdrawn position, and then the removal of the blade guide and the fuel support casting.

It is possible to disconnect the control rod from its control rod drive from the open vessel or from the under side of the vessel. When the control rod is uncoupled from inside the open vessel, the control rod latch tool is used. When uncoupling from the under side of the vessel by means of disconnecting mechanism that is part of the control rod drive, an uncoupling tool is used. The rods are generally uncoupled from the under side of the vessel. Once the control blade is uncoupled, it is removed to the fuel transfer pool for transfer to the control rod storage rack in the fuel handling area using a control rod grapple attached to one of the refueling platform auxiliary hoists. The servicing of vessel internals is accomplished from either the refueling platform or from the vessel service platform, shown in Figure 12.3-17. The reactor vessel service platform is a motor driven structural steel assembly designed to provide close access to the vessel internals.

The platform rides on a self contained rail which also serves as its lifting strongback. This rail attaches to and is supported by the reactor vessel flange. Located on the service platform is a jib crane mounting socket, a work well, and several smaller openings through which various vessel internal components are serviced.

The vessel internals are serviced with the aid of assorted tools and accessories which are operated from the refueling platform and/or from the vessel service platform. The use of these tools and accessories is not restricted to vessel servicing; many of them are also used in the fuel handling area pools. The primary function of the under reactor vessel servicing equipment is to remove and install control rod drives, service thermal sleeves and control rod guide tubes, and install and remove the neutron detectors.

The control rod drive handling equipment is powered pneumatically and is designed for the removal and installation of the control rod drives from their housings. This equipment is used in conjunction with the equipment handling platform, shown in Figure 12.3-18. The equipment handling platform is located directly below the reactor pressure vessel. The platform is capable of rotating forward and reverse for 360 degrees with access to such items as the control rod drives, source range and intermediate range monitor drives, electrical connections for the incore detector assemblies, and the tubing and connectors for the traversing incore probes.

The control rod drive handling equipment, in conjunction with the equipment handling platform, is used to remove a control rod drive from the reactor and transport it out of the drywell to a service and repair area. When the reactor is shut down, cooled down, and depressurized, the control rod to be removed is fully withdrawn and uncoupled from its drive, and the position indication probe and CRD housings supports are removed.

With the transfer cart removed from the equipment handling platform and placed in the leveling tray outside the under vessel service area, the handling platform is rotated so that the centerline of the platform opening aligns with the CRD to be removed. After the CRD transfer cart has been properly loaded, it is pulled into the containment and transported to the CRD storage and repair room using a winch outside the drywell.

Local power range detector dry tubes are removed from or installed in the pressure vessel at a time when the four fuel bundles in proximity to the detectors have been removed. The local power range detector assembly consists of a multiple dry tube cluster with four neutron detectors inside them, all located in the water gap between four fuel bundle groups that are enclosed by four control rods. The individual detectors are withdrawn from the dry tubes from beneath the vessel and fed into a shielded disposal cask and cut into sections. The cask is then moved to the fuel handling area storage pool to await further disposal of contents.

Replacement detectors are installed from underneath the vessel by inserting them manually into the dry tubes until the fitting on the detector mates with the fitting on the dry tube. When the fitting is tightened, the mechanical installation is complete. The signal cable is connected to the electrical connection of the detector and the replacement is ready for test.

The critical path fuel handling operations in the fuel handling area are performed by the spent fuel gantry crane. Therefore, the operation of the new fuel bridge crane is coordinated with the operations of the spent fuel gantry crane to ensure a safe, continuous, fuel handling process.

When refueling and servicing are completed, the vessel internals and vessel and drywell heads are replaced. The reactor well is again filled. When all material is transferred from the upper containment pool to the fuel handling area, the isolation gate-valves are closed and startup operations can begin.

12.3.4 System Features and Interrelations

The special features of this system have already been covered in the component description section. The interrelations between this system and other plant systems are discussed in the paragraphs that follow.

12.3.4.1 Fuel Transfer System (Section 12.2)

The Fuel Transfer System is used to transfer fuel, control rods, and other small items between the fuel building and the containment.

12.3.4.2 Rod Control and Information System (Section 7.1)

The Rod Control and Information System (RC&IS) provides rod withdrawal blocks and interlocks when certain Refueling and Vessel Servicing System equipment is in use. The detailed description of these rod withdrawal blocks is given in table 7.1-12 in Section 7.1.

12.3.5 BWR Differences

The discussion in this section is typical for facilities of product lines BWR/2 through BWR/6. Minor differences may exist from plant to plant, depending on the containment design.

12.3.6 Summary

Classification - Power generation system

Purpose - To provide facilities for the handling and storage of new and spent fuel and to provide equipment for vessel refueling and servicing of reactor internal components

Components - Fuel Handling and Storage Subsystem, Fuel Servicing Subsystem, Vessel Servicing Subsystem

System Interrelations - Fuel Transfer System, Rod Control and Information System.

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Chapter 13.0

Reactor Operations

13.0 REACTOR OPERATIONS

Detailed written operating procedures for all modes of plant operation are prepared prior to the initial startup and critical testing period. Appropriate changes in these procedures are made during the startup test program. The following is a discussion of the general operating procedures that are used for plant startup, power operation, and shutdown. This information is presented to indicate the general method of operation.

The order to startup the reactor for power operation, or to shutdown for maintenance or refueling, is issued by the plant superintendent or assistant superintendent. The load schedule during power operation is issued by the load dispatcher's office with concurrence from the plant. To support these operations, the shift supervisor schedules the startup or shutdown of various systems and components required, issuing the necessary orders to the control room personnel.

The following sections describe the sequence of general plant operations during startup, power operation, and shutdown.

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Chapter 13.2

Power Operations

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13.2 POWER OPERATION

After the generator is synchronized to the utilitys' transmission grid and is producing a substantial output, reactor power output is varied to meet the grid system requirements by adjustment of control rod position, manual adjustment of reactor recirculation flow, or a combination of these two methods.

13.2.1 Control Rod Adjustment

Withdrawing a control rod reduces the neutron absorption and adds positive reactivity to the core. Reactor power then increases until the increased steam formation just balances the change in reactivity caused by the rod withdrawal. The increase in boiling rate tends to raise reactor pressure, causing the EHC System pressure regulator to open the turbine control valves sufficiently to maintain a programmed throttle pressure. When a control rod is inserted, the reverse effect occurs.

13.2.2 Recirculation Flow Control

Reactor power output can be varied over a power range of approximately 35% of rated power by adjustment of reactor recirculation flow, while maintaining a nearly uniform power distribution. Reactor power change is accomplished by using the negative void coefficient. An increase in recirculation flow temporarily reduces the volume of steam in the core by raising the boiling boundary. This addition of reactivity in the core causes reactor power level to increase. The resultant increased steam generation rate then returns the steam volume in the core to approximately its original value, and a new constant power level is established. When recirculation flow is reduced, the power level is reduced in a similar manner.

During initial power operation, the operating curve or power/flow map is established relating reactor power to recirculation flow. The first point of the curve is full flow and rated power. When a rod pattern is established for this point, recirculation flow is reduced in steps at the same rod pattern, and the relationship of flow to power is plotted for steady state conditions. Other curves are established at lower power ratings and other rod patterns as desired. During operation, reactor power may be changed by flow control adjustment, rod positioning, or a combination of the two, while adhering to established operating curves.

Although control rod movement is not required when the load is changed by recirculation flow adjustment, the long term reactivity effects of fuel burnup is compensated for by control rod adjustment.

Operating personnel are engaged in many activities during normal power operation. The following list includes some of the most typical activities:

1. Routine data taking and writing of operational logs.

2. Routine system instrument and valve tests.

3. Special periodic operational tests.

4. Manipulation of control rods and recirculation flow to maintain a balanced flux distribution, accommodate major changes in load demand, and secure optimum plant performance.

5. Evaluation of abnormal conditions as annunciated and indicated, and taking required action to minimize potentially dangerous effects on equipment and systems.

6. Sampling of process steam and water.

7. Surveillance of plant equipment for proper operation, including performing necessary adjustments and minor repairs.

8. Confining radioactive contamination to the smallest possible area, and preventing contamination of personnel areas and the environment.

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Chapter 13.3

Normal Shutdown from Power

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Control rods are withdrawn to no greater than the 80% rod line to prevent entering the instability region. At this time reactor power is increased by increasing core flow by the RFC system. Core flow could have been increased as soon as the 20% feedwater interlock in the RFC system had been satisfied. Core flow is increased to greater than 35 Mlbm/hr. then control rods are withdrawn to establish the 100% control rod pattern. When reactor recirculation speed is at or greater than 45% the RFC system is placed in Master Manual. Core flow is now increased to 100% of rated or to 100% rated thermal power(2436 MWth) whichever occurs first.

Twenty four hours after reaching 15% of rated power the Primary Containment oxygen concentration must be 4% or less. Core thermal limits and APRM calibrations are performed during power ascension.

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Appendix A

Glossary Of Terms

This Glossary of Terms is taken in part from NUREG-0770, "Glossary of Terms: Nuclear Power and Radiation."

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Appendix A

Glossary Of Terms

This Glossary of Terms is taken in part from NUREG-0770, "Glossary of Terms: Nuclear Power and Radiation."

GLOSSARY OF TERMS*
(FOR BOILING WATER REACTORS)

TERM	DEFINITION
absorber tubes	The tubes that form the control rods contain natural boron in the form of B_4C. The n-a reaction produces Li-7 and He-4.
access lock	A lock that preserves the containment's pressure integrity while transferring equipment or personnel through the containment.
accident analysis	A standard approach to ensure the license applicant has analyzed both expected and unexpected transients and accidents, that conservative values are applied in the analysis,and that the plant design has incorporated an adequate margin of safety. The accident analysis section of the FSAR shall include the analysis of several transients and accidents. These analyzed incidents are placed in four categories or conditions of operation; normal operations, faults of moderate frequency, infrequent faults, and limiting faults.
active failure	A malfunction, excluding passive failures, of a component which relies on mechanical movement to complete its intended function on demand.
active fuel length	End-to-end dimension of fuel material within a fuel element.
adsorption	Retention of materials by adhesion to the surface of another.
airborne radioactive	Any radioactive material dispersed in the air in the material form of dusts, fumes, mists, vapors, or gases.
alpha radiation (α)	A positively charged particle emitted by certain radioactive materials. It is composed of two neutrons and two protons, hence is identical with the nucleus of a helium atom. It is the least penetrating of the three common types of radiation (alpha, beta, gamma) emitted by radioactive material. It is stopped by a sheet of paper. It is not dangerous to plants, animals, or man, unless the emitting substance has entered the body.

*These are the more commonly used terms. This list is not intended to be all inclusive.

average planar linear heat generation rate (APLHGR)	The average value of the linear heat generation rate of all the fuel rods at any given horizontal plane along a fuel bundle.
axial peaking factor (APF)	The ratio of the heat flux at the axial location of interest to the heat flux average over the active length of the fuel (assembly or rod) of interest.
axial power shape	The relative heat flux along an axial line. In each core region, the axial shape can be determined by a TIP traverse or by fitting a curve through a set of LPRM values. Knowledge and control of axial shape throughout the core is aided by maintaining symmetry in the rod pattern and a similar axial shape in all radial regions.
base loaded	A term describing a power plant which is maintained at constant load. A base-loaded plant is the opposite of a load-following plant.
beta radiation (ß)	An elementary particle emitted from a nucleus during active decay, with a single electrical charge and a mass equal to 1/1837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is called a positron. Beta radiation may cause skin burns, and beta-emitters are harmful if they enter the body. Beta particles are easily stopped by a thin sheet of metal.
biological shield	A mass of absorbing material placed around a reactor or radioactive source to reduce the radiation to a level that is safe for human beings.
black and white rod pattern	One in which each rod is either fully inserted (black) or fully withdrawn (white) and no rods are partially withdrawn (gray).
boiling	The process in which vapor forms within a continuous liquid phase. Typically, tiny bubbles of vapor which grow and detach are formed at the heat-transfer surface. This is <u>nucleate boiling</u> which may be subcooled or saturated. <u>Subcooled boiling</u> occurs when the bulk liquid temperature is below saturation, and only the liquid film at the heat transfer surface is at saturation. <u>Film boiling</u> occurs when a continuous film of vapor blankets the heat transfer surface. The transition from subcooled to saturated nucleate boiling is continuous; the transition to film boiling is relatively sharp.
boiling length (LB)	The linear distance from the onset of bulk boiling to the transition boiling point.

boiling water reactor — A power reactor in which water, used as both coolant and moderator, is allowed to boil in the core. The resulting steam can be separated from the water and fed either directly or through a heat exchanger to a turbine-generator.

burnable poison — A neutron absorber or poison (such as gadolinium) that, when purposely incorporated in the fuel or fuel clad- ding of a nuclear reactor, gradually "burns up" (is changed into nonabsorbing material) under neutron irradiation. This process compensates for the loss of reactivity that occurs when fuel is consumed and fission- product poisons accumulate, and keeps the overall characteristics of the reactor nearly constant.

capacity factor — The ratio of the average power load of an electric power plant to its rated capacity. Sometimes called "plant factor."

carryover — The weight-fraction of liquid water carried as droplets with the steam as it leaves the reactor. Water droplets cause erosion in the turbine and carry activated corrosion products and fission products into the turbine, potentially causing maintenance problems. Carryover to the turbine is typically kept below 0.1 wt%.

carryunder — The weight-fraction of steam bubbles entrained in the recirculating reactor water, i.e., not separated out and sent to the turbine. The entrained steam reduces the subcooling at the reactor inlet and thus increases the local and exit steam qualities of the fuel assemblies and core average voids. It may also aggravate or cause recirculation or jet pump cavitation. Carryunder is principally a drag on plant efficiency, whereas carryover is a hazard to the turbine.

cavitation — Bubble formation and collapse at a low-pressure point in a flowing stream. Bubbles will form where the local pressure falls below the vapor pressure of the liquid and will then be transported with the liquid, and collapse where the pressure is above the vapor pressure. This is a common potential problem with large pumps when operated at "off-standard" conditions.

cerenkov radiation — Light emitted when charged particles pass through a transparent material at a velocity greater than that of light in that material. It can be seen, for example, as a blue glow in the water around the fuel elements of pool reactors. P. A. Cerenkov is the Russian scientist who first explained the origin of this light.

coast down	The stretch out of a cycle by holding a constant control rod pattern and permitting the reactor power level to decrease gradually as the core reactivity decreases.
contamination, radioactive	Deposition of radioactive material in any place where it is not desired, particularly where its presence may be harmful.
control rod withdrawal sequence	The order in which control rods are scheduled to be withdrawn. The sequence begins with all rods inserted and is, or can be, extended to withdrawal of all rods.
control rod worth	The reactivity worth of a single control rod in a finite reactor core under a defined set of conditions.
cooldown	The cooling of a reactor after it has been shut down.
core flow	That coolant going through the core. See "jet pump flow."
critical	Capable of sustaining a chain reaction at a constant level. Prompt critical is being capable of sustaining chain reaction without the aid of delayed neutrons.
critical power (CP)	The fuel bundle power, above which, the nucleate boiling process breaks down at some point within the bundle and transition boiling commences. It is characterized by abrupt, unstable variations in heat transfer surface temperature, and is a function of inlet enthalpy, bundle steam quality, boiling length, etc.
critical power ratio (CPR)	The ratio of critical power to bundle operating power; used as a figure of merit to evaluate BWR core thermal performance.
criticality	The state of a nuclear reactor when it is sustaining a chain reaction.
daughter	A nuclide formed by the radioactive decay of another nuclide, called in this context the parent.
decay heat	The heat produced by the decay of radioactive nuclides. Decay heat is released in a reactor following shutdown, first from fissions caused by delayed neutrons and heat capacity of core components and, over a longer period, by the radioactive decay of fission products in the reactor. This requires provision for cooling a reactor for long periods of time following shutdown.

decontamination	The removal of radioactive contaminants from surfaces or equipment, as by cleaning and washing with chemicals.
derate	Any administrative action which limits the reactor power to a value less than the nameplate rating. Derating can be used for cycle stretch-out. It may or may not involve a physical limitation in plant capability.
detergent waste filter	The detergent waste filter removes lint and other particulate matter from the radioactive decontamination drains. The filter is equipped with inlet and outlet connections in addition to vent and drain connections.
doppler effect	An increase in neutron absorption by a material as a result of an increase in its temperature. The Doppler effect of the fertile material in a reactor is an important factor for achieving safety in large thermal and fast reactors because of the decrease in reactivity with temperature.
doppler (fuel temperature coefficient)	The change in the core reactivity level for a unit change in the fuel temperature. The change results from the broadening effects of temperature on the neutron absorption resonance.
doubling time	The time for the neutron flux level to double.
driving flow	Driving flow (sometimes called recirculation flow) measurements are commonly made of the DP across the pump(s) or venturi. This is the flow through the recirculation pumps.
dryer/separator canal	The dryer and separator canal provide underwater access between the dryer-separator storage pool and the reactor well.
dryer/separator storage pool	The pool is located on the refueling floor and provides storage and servicing facilities for the dryer and separator when they have been removed from the reactor vessel.
drywell	The containment vessel enclosing the reactor and recirculation system and forming part of the primary pressure suppression system.
electrical capacity factor	The ratio of electrical energy produced in a given time interval to the electrical energy that would have been produced in that same interval if the turbine-generator were to operate continuously as its maximum rating. Note that the thermal capacity factor and the electrical capacity factor would, in general, be different and that the thermal capacity factor is the term that is more generally useful in any problem involving fuel

exposure.

elevation head	Pressure exerted by a column of fluid which is proportional to the density and height of the column. (The difference in elevation head of the boiling fuel channels and a subcooled annular region provides the pressure driving head of natural circulation reactors.)
enrichment	A process by which the relative abundances of the isotopes of a given element are altered, thus producing a form of the element that has been enriched in one particular isotope.
enthalpy	Internal energy plus mechanical equivalent of heat energy contained by a unit mass of fluid.
excess reactivity	More reactivity than that needed to achieve criticality. Excess reactivity is built into a reactor (by using extra fuel) in order to compensate for fuel burnup and the accumulation of fission-product poisons during operations.
exclusion area	An area immediately surrounding a nuclear reactor where human habitation is prohibited to ensure safety in the event of an accident.
exit quality	Quality existing at the effluent end of a fuel channel or of the entire core.
fast fission	Fission resulting from the collision of a nucleus and a high-energy neutron. Some nuclei, such as those of U-238, fission only by fast neutrons.
fertile	Capable of being transformed into a fissionable substance by capture of a neutron. Fertile material, not itself fissionable by thermal neutrons, can be converted into a fissile material by irradiation in the reactor. Common examples are U-238, Th-232 and Pu-240. When these fertile materials capture neutrons, they are partially converted into fissile Pu-239, U-233 and Pu-241, respectively.
film boiling	See "boiling"
filter/demineralizer	Removes chlorides, sulfides, oxides, etc. from radwaste effluent using Powdex resins or other suitable filtration material.
fission fragments	The two nuclei that are formed by the fission of a nucleus. Also referred to as primary fission products. They have medium atomic weights and are radioactive.

fission gas	Those fission products which exist in the gaseous state at normal temperatures and pressures.
fission, nuclear	The division of a heavy nucleus into two approximately equal parts. For the heaviest nuclei the reaction is highly exothermic, the release of energy being about 210 MeV per fission. A well-known example is the spontaneous fission of U-238. Other examples are the fissions of U-233 and Pu-239 after neutron capture.
fission poisons	Fission fragments that readily absorb neutrons; for example, Xe-135, which has an absorption cross section of 3.5 million barns for slow neutrons.
fission-product poisoning	The absorption or capture of neutrons by fission products in a reactor, thereby decreasing its reactivity.
fission products	The nuclei (fission fragments) formed by the fission of heavy elements, plus the nuclides formed by the fission fragments' radioactive decay. Some of the fission products are, or become, strong neutron absorbers.
fissionable	Capable of being fissioned by the capture of a particle, such as a neutron or photon. The most common fissionable materials are U-235, Pu-239, and U-233.
flux; neutron flux (see also "heat flux")	A measure of the intensity of neutron radiation. It is the number of neutrons passing through 1 square centimeter of a given target in 1 second. Expressed as nv, where n = the number of neutrons per cubic centimeter and v = their velocity in centimeters per second.
forced circulation	Forced circulation is induced partially by rotating and partially by jet pumps. The gross core flow and the individual fuel assembly flows are weak functions of power.
fuel assembly	A fuel assembly is a "bundle" of fuel rods held in a rigid rectangular array by tie-plates at the top and bottom, supported at intermediate levels with spacers, and enclosed by a fuel channel.
fuel bundle sampler	The device for obtaining water or gas samples from a fuel bundle in the shutdown reactor or in the storage pool. (Also called the "sipper.")
fuel storage pool	A pool that provides storage and servicing facilities for activated fuel

elements.

fuel time-constant — The specific heat of the UO-2, combined with its thermal conductivity, yields a fuel temperature time-constant of a few seconds. The time-constant is the time required for the temperature to change by a fraction 1/e of the steady-state temperature differences associated with two flux levels if the flux level change is made as a step. This few-second fuel time constant is a very important characteristic of all UO_2 fueled reactors.

gamma radiation (γ) — High-energy, short-wavelength, electromagnetic radiation.Gamma radiation frequently accompanies alpha and beta emissions and always accompanies fission. Gamma rays are very penetrating and are best stopped or shielded against by dense materials, such as lead or depleted uranium. The rays are similar to X rays, but usually are more energetic, and are nuclear in origin.

half-life — The time in which half the atoms of a particular radioactive substance disintegrate to another nuclear form. Each radionuclide has a unique half-life. Measured half-lives vary from millionths of a second to billions of years.

heat flux — Rate of heat flow across a boundary (usually cladding surface), expressed in Btu/hr-ft or watts/cm^2.

hertz (Hz) — Unit of frequency, equal to one cycle per second.

hot — A slang term meaning highly radioactive.

hot spot — A surface area of higher-than-average radioactivity. Also a part of a fuel element surface that has one of the highest heat fluxes in the core.

important to safety — Those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Encompasses the broad class of plant features that contribute in an important way to the safe operation and protection of the public in all phases and aspects of facility operations. Includes Safety Grade or Safety-Related as a subset.

instrument channel — An arrangement of sensor and associated components used to evaluate plant variables and produce discrete outputs used in logics. A channel terminates and loses its identity where individual channel outputs are combined in logics.

integrated neutron flux	Flux multiplied by time; usually expressed as nvt, where n = the number of neutrons per cubic centimeter, v = their velocity in centimeters per second, and t = time in seconds.
interlock	A device usually electrical and/or mechanical, to prevent activation of a control until a preliminary condition has been met, or prevent hazardous operations. Its purpose usually is safety.
ion exchange	A chemical process involving the reversible interchange of various ions between a solution and solid material, usually a plastic or resin. The process is used to separate and purify chemicals, such as fission products and rare earths, in solutions.
ionization	The process of creating ions by adding or subtracting one or more electrons to or from atoms or molecules. High temperatures, electrical discharges, or nuclear radiation can cause ionization.
irradiation	Exposure to radiation, as in a nuclear reactor.
isolated condition	Normal isolation of the reactor from the main condenser, including the closure of the main steam line isolation valves.
jet pump flow	The same (except for the relatively insignificant control rod drive flow) as total core flow. The driving flow, or recirculation flow, mixes with suction flow in the throat and diffuser of the jet pump before entering the core.
leakage, neutron	The loss of neutrons from a reactor core by outward diffusion. When there is a reflector, leakage refers to net loss of neutrons that leave the core and are not reflected back into it. Leakage lowers the neutron level in a reactor.
leakage flow	Coolant flow that is diverted to other regions of the reactor outside the fuel channels to remove heat from control rods, sources and fission chambers. About 10% of the total core flow is leakage flow, removing about 3% of the heat generated in the core.
limiting conditions for operation (LCO)	Specify the minimum acceptable levels of system performance necessary to ensure safe startup and operation of the facility. When the conditions are met the plant can be operated safely and abnormal situations can be safely controlled.
limiting safety system	Instrumentation settings which initiate automatic protective action

setting (LSSS) — at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings.

linear heat generation rate (LHGR) — The heat generation system rate per unit length of a fuel rod. Common units are kW/ft.

load following — A term describing a power plant whose power is raised and lowered to meet the day-to-day demands of its electrical grid. A load-following plants the opposite of a base-loaded plant.

local peaking factor — Ratio of the maximum-to-average fuel rod power within a fuel assembly.

local power — The power generation in an arbitrary unit of volume, usually a small length of a fuel assembly called a node. It is the integral of the heat flux over the heat transfer area in the unit of volume or length, plus an increment for the heat deposited in the water by thermalization of neutrons and absorption of gamma energy where applicable.

logic — That array of components which combines individual bistable output signals to produce decision outputs.

maximum average planar linear heat generation rate (MAPLHGR) — The maximum in-core value of average planar linear Heat generation rate.

megawatt-day per ton — A unit for expressing the burnup of fuel in a reactor; specifically, the number of megawatt-days of heat per metric ton.

minimum critical power ratio (MCPR) — The smallest critical power ratio existing anywhere in the core. This expression is used in place of such terms as "minimum burnout ratio" and "minimum burnout margin." The control room problem is to determine the magnitude, and the location in the core, of the MCPR. Nomograms, worksheets, and procedures are provided so that the MCPR value can be determined for the operating condition that exists. An on-line computer may be used to determine this important value. Ordinarily, it is sufficient to determine the MCPR value following any change in operating conditions, or at intervals of once per shift of base load operation.

moderator — A material used in a reactor to slow down high-velocity neutrons and increase the likelihood of further fission. Moderators commonly include ordinary water, heavy water and graphite. Liquid moderators can also serve as the coolant. Neutrons lose energy by scattering collisions with nuclei of the moderator. A good moderator has high scattering cross section and low atomic weight.

moderator temperature coefficient	The change in the core reactivity level per a unit change in the moderator temperature. The moderator temperature coefficient of reactivity is the composite of three principal effects. These are: (1) the temperature effect in k^{∞}, (2) the temperature effect on core neutron leakage, and (3) the temperature effect on the control rod system worth. The latter is a large negative effect. The composite coefficient becomes less negative with fuel depletion, reaching the least negative value at the end of each fuel cycle.
moderator void coefficient	The change in the core reactivity level per a unit change in the moderator void content. The moderator void coefficient of reactivity is a composite of the same three effects as in the temperature coefficient, but refers only to in-channel changes in moderator density, i.e., voids. As in the temperature coefficient, the void coefficient becomes less negative with fuel depletion.
multiplication factor	The effective multiplication constant (K_{eff}) is the ratio of the number of neutrons present in a reactor at a given time to the number present one finite neutron lifetime earlier. The "excess reactivity" is K_{eff}-1 which can be likened to the "interest rate per neutron lifetime," and excess of 1% means that the neutron population (capital) will increase by a factor of 1.01 in a one neutron lifetime. A reactor is said to be subcritical if $K_{eff}<1$, critical if $K_{eff}=1$, and supercritical if $K_{eff}>1$.
natural circulation	The coolant (usually water) in a reactor is circulated without pumping; that is, by natural convection resulting from the different densities of relative cold and heated portions.
natural uranium	Uranium as found in nature, containing 0.7% U-235, 99.3% U-238, and a trace of U-234.
net positive suction head (NPSH)	The pressure head, in feet of fluid, acting to suppress cavitation at the elevation of interest. It is the excess of the static head over the saturation pressure corresponding to the fluid temperature. The NPSH can be increased by greater submergence of the pump below the fluid surface and/or by greater subcooling of the suction fluid.
neutron flux	See "flux."

neutron leakage	See "leakage."
noble gases	Radioactive (nonreactive) elements, such as argon, xenon, and krypton released in the fission process.
nuclear fission	See "fission, Nuclear."
nucleate boiling	See "boiling."
nv	Neutron flux, units of neutrons per second per square centimeter.
offgas	The accumulation of air through in-leakage around the BWR turbine, the fission gases present in the steam and the hydrogen and oxygen from disassociation of water; and exhausted through the steam jet air ejectors or mechanical vacuum pumps.
passive failure	A breech of a fluid pressure boundary or blockage of a process flow path.
peaking factor, total	The ratio of the maximum fuel rod surface heat flux in any assembly to the core average fuel rod surface heat flux.
plant factor	See "capacity factor."
period	The time required for one cycle of a regularly repeated series of events. In a nuclear reactor, it is the time required for the power level to change by the factor of 2.718, which is known as e (the base of natural logarithms).
poison	Any material of high absorption cross section that absorbs neutrons unproductively and hence removes them from the fission chain reaction in a reactor and decreases its reactivity. It may be in the form of poison curtains or an accumulation of material resulting from operation, especially fission products of high-neutron-absorption cross section.
power	<ins>Power</ins> is the rate of heat production, transfer or flow. <ins>Fission power</ins> or (neutron power) refers to the rate of the basic fission process, and this responds essentially instantaneously to a neutron flux change. Fission power can be determined only by calibrated nuclear instrumentation during a fast transient. <ins>Heat flux power</ins> refers to the rate of transfer of the heat from the fuel to

the coolant. In any neutron flux transient, the rate of heat transfer lags behind the fission power generation rate because of the few second time constant of the UO2 fuel. Since heat flux cannot be measured directly, transient values must be calculated from other measurements.

Reactor power is determined by a coolant heat-balance under steady-state conditions by measuring pressure, or temperatures, and coolant flows to determine enthalpys and heat removal and addition rates. The calibration of primary nuclear instrumentation and the basic plant heat production records depend directly on the heat-balance data.

power density — The rate of heat generated per unit volume of a reactor core.

quality, steam — The percentage of weight of water that is in the vapor phase. In fluid flow, steam quality refers to the weight-fraction of steam in the two-phase flowing mixture of steam and liquid in the channel passing the point of interest. Thus, in a fuel assembly the quality increases from zero (below the elevation where boiling begins) to the maximum or exit value at the top of the assembly. Steam quality is the important property when considering the thermal properties of the fluid.

radial peaking factor (RPF) — The ratio of the fuel assembly power or heat flux in a particular assembly to the power or heat flux of the core average fuel assembly.

radiation — The emission and propagation of energy through matter or space by means of electromagnetic disturbances that display both wave like and particle like behavior. The term radiation; such radiation is commonly classified according to frequency, as Hertzian, infrared, visible (light), ultraviolet, X-ray, and gamma-ray. By extension, corpuscular emissions, such as alpha and beta radiation, or rays of mixed or unknown type, are classified as cosmic radiation. Nuclear radiation is that emitted from atomic nuclei in various nuclear reactions, including alpha, beta, and gamma radiation and neutrons.

reactivity — A measure of the departure of a nuclear reactor from criticality. It is about equal to the effective multiplication factor minus one and is thus precisely zero at criticality. If there is excess reactivity (positive reactivity), the reactor is supercritical and its power will rise. Negative reactivity (subcriticality) will result in a decreasing power level.

recirculation flow — See "driving flow."

refueling — The removal and addition of fuel assemblies to the core; however, the term is often extended to include any and all additions, rearrangements,

	or removals of core components which affect reactivity.
refueling platform	The platform that moves over the reactor pool and the fuel storage pool to carry operators, refueling tools, and fuel.
refueling outage	Includes all of the planned operations associated with a normal refueling except those tests in which the reactor is taken out of and returned to the shutdown (more than one rod subcritical) condition. The following operations are included in refueling: planned physical movement of core components (fuel, control rods, etc.), refueling test operations (except criticality and shutdown margin tests), and planned maintenance.
roentgen equivalent man (rem)	A measure of the dose of any ionizing radiation to body tissues in terms of its estimated biological effect. A dose of 1 rad of X, gamma or ß radiation is equivalent to 1 rem.
safety-grade	This term is not explicitly used in regulations. It is equivalent to "Safety-Related", and is a subset of "Important to Safety".
safety limits	Limits below which the reasonable maintenance of the integrity of the cladding and primary systems are assured. Operation beyond such a limit may not in itself result in serious consequences, but indicate an operational deficiency subject to regulatory review.
safety related	Those structures, systems, or components designed to remain functional for the Safe Shutdown Earthquake necessary to assure required safety functions. i.e: (1) The integrity of the reactor coolant system pressure boundary (2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guidelines in 10 CFR 100 Appendix A.
saturation	Refers to the enthalpy or temperature of a liquid at which the vapor pressure equals the local pressure. At saturation, further additions of heat cause some of the liquid to change to vapor; that is, boiling occurs.
scram	The sudden shutdown of a nuclear reactor by rapid insertion of the control rods. Emergencies or deviations from normal reactor operation cause the reactor operator or automatic control equipment to scram the reactor.
secondary containment	This is attained when the reactor building is closed and the following

integrity	conditions are met: At least one door at each access opening is closed. The Standby Gas Treatment System is operable. All reactor building ventilation system automatic isolation valves are operable or are secured in the closed position.
separator/dryer storage pool	The pool is located in the refueling floor and provides storage and servicing facilities for the separator and dryer when they are removed from the reactor vessel.
service platform	The platform placed over the reactor at the level of the vessel flange to permit operators to work on the core. Normally, this platform rests on a seal surface protector.
shutdown margin	The amount of reactivity by which the reactor is subcritical. Mathematically, $1-K_{eff}$ (for $K_{eff} < 1$). The value specified normally assumes that the strongest control rod is stuck in the fully withdrawn condition.
single failure	An occurrence which results in the loss of capability of a component to perform its intended safety function when called upon. Multiple failures resulting from a single occurrence are to be considered a single failure. Fluid process systems are considered to be designed against an assumed single failure if neither a single active nor a single passive failure results in a loss of the safety function to the nuclear unit.
sipper	See "fuel bundle sampler."
spent (depleted) fuel	Nuclear reactor fuel that has been irradiated to the extent that it can no longer effectively sustain a chain reaction.Fuel becomes spent when its fissionable isotopes have been partially consumed and fission - product poisons have accumulated in it.
stack	The chimney used to disperse the offgas from reactor operation.
storage and handling facilities, fuel	Fuel storage and handling facilities and procedures are designed to ensure that an unintentional criticality cannot occur. Any proposed change in equipment, geometry, or procedure involving either new or exposed fuel must be rigorously examined for a change in criticality risk.
subcooled boiling	See "boiling."
subcooling	The difference between the saturation enthalpy and the actual enthalpy of water in the liquid phase. Common units are Btu/lb, cal/gm, and ^{0}F.

suction flow	Suction flow is a mixture of subcooled feedwater and saturated water discarded by the steam separators and dryers above the jet pump suction inlet. This suction flow is drawn into the jet pump by the venturi effect of the driving flow.
superheating	A heating of a vapor, particularly saturated (wet) steam, to a temperature higher than the boiling point at the existing pressure. This is done in power plants to improve efficiency and to reduce condensation in the turbines.
temperature coefficient of reactivity	The change in reactor reactivity (per degree of temperature) occurring when the operating temperature changes. The coefficient is positive when an increase in temperature increases the reactivity and negative when an increase in temperature decreases reactivity. Negative temperature coefficients help to prevent power excursions. See"Doppler (Fuel Temperature) Coefficient" and "Moderator Temperature Coefficient."
thermal efficiency	The ratio of the electric power produced by a power plant to the amount of heat produced by the fuel; a measure of the efficiency with which the plant converts thermal energy to electrical energy.
thermal (slow) neutron	A neutron in thermal equilibrium with its surrounding medium. Thermal neutrons are those that have been slowed down by a moderator to an average speed of about 2200 meters per second (at room temperature) from the much higher initial speeds they had when expelled by fission. This velocity is similar to that of gas molecules at ordinary temperatures.
trip system	An arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function.
tripped (technical specifications)	The change of state of a bistable device which represents the change from a normal condition. A trip signal, which results from a trip, is generated in the channels of a trip system and produces subsequent trips and trip signals through the system as directed by the logic.
two-phase pressure drop	The DP produced by the flow of a mixture of liquid and vapor through a resistive path. The DP for a given mass flow rate is dependent on the weight fraction of the mass flow that is steam.
velocity limiter	An integral part of the control rod designed to limit the free-fall velocity

	of the control rod in water.
void coefficient	The change in reactivity resulting from a percentage change in void fraction.
void fraction	Fraction of the volume of the coolant stream (moderator) that is in the vapor phase. Void fraction is an important property when considering nuclear properties such as reactivity, moderation, etc.

GLOSSARY OF TERMS*
(FOR BOILING WATER REACTORS)

TERM	DEFINITION
absorber tubes	The tubes that form the control rods contain natural boron in the form of B_4C. The n-a reaction produces Li-7 and He-4.
access lock	A lock that preserves the containment's pressure integrity while transferring equipment or personnel through the containment.
accident analysis	A standard approach to ensure the license applicant has analyzed both expected and unexpected transients and accidents, that conservative values are applied in the analysis,and that the plant design has incorporated an adequate margin of safety. The accident analysis section of the FSAR shall include the analysis of several transients and accidents. These analyzed incidents are placed in four categories or conditions of operation; normal operations, faults of moderate frequency, infrequent faults, and limiting faults.
active failure	A malfunction, excluding passive failures, of a component which relies on mechanical movement to complete its intended function on demand.
active fuel length	End-to-end dimension of fuel material within a fuel element.
adsorption	Retention of materials by adhesion to the surface of another.
airborne radioactive	Any radioactive material dispersed in the air in the material form of dusts, fumes, mists, vapors, or gases.
alpha radiation (α)	A positively charged particle emitted by certain radioactive materials. It is composed of two neutrons and two protons, hence is identical with the nucleus of a helium atom. It is the least penetrating of the three common types of radiation (alpha, beta, gamma) emitted by radioactive material. It is stopped by a sheet of paper. It is not dangerous to plants, animals, or man, unless the emitting substance has entered the body.

*These are the more commonly used terms. This list is not intended to be all inclusive.

average planar linear heat generation rate (APLHGR)	The average value of the linear heat generation rate of all the fuel rods at any given horizontal plane along a fuel bundle.
axial peaking factor (APF)	The ratio of the heat flux at the axial location of interest to the heat flux average over the active length of the fuel (assembly or rod) of interest.
axial power shape	The relative heat flux along an axial line. In each core region, the axial shape can be determined by a TIP traverse or by fitting a curve through a set of LPRM values. Knowledge and control of axial shape throughout the core is aided by maintaining symmetry in the rod pattern and a similar axial shape in all radial regions.
base loaded	A term describing a power plant which is maintained at constant load. A base-loaded plant is the opposite of a load-following plant.
beta radiation (ß)	An elementary particle emitted from a nucleus during active decay, with a single electrical charge and a mass equal to 1/1837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is called a positron. Beta radiation may cause skin burns, and beta-emitters are harmful if they enter the body. Beta particles are easily stopped by a thin sheet of metal.
biological shield	A mass of absorbing material placed around a reactor or radioactive source to reduce the radiation to a level that is safe for human beings.
black and white rod pattern	One in which each rod is either fully inserted (black) or fully withdrawn (white) and no rods are partially withdrawn (gray).
boiling	The process in which vapor forms within a continuous liquid phase. Typically, tiny bubbles of vapor which grow and detach are formed at the heat-transfer surface. This is <u>nucleate boiling</u> which may be subcooled or saturated. <u>Subcooled boiling</u> occurs when the bulk liquid temperature is below saturation, and only the liquid film at the heat transfer surface is at saturation. <u>Film boiling</u> occurs when a continuous film of vapor blankets the heat transfer surface. The transition from subcooled to saturated nucleate boiling is continuous; the transition to film boiling is relatively sharp.
boiling length (LB)	The linear distance from the onset of bulk boiling to the transition boiling point.

boiling water reactor

A power reactor in which water, used as both coolant and moderator, is allowed to boil in the core. The resulting steam can be separated from the water and fed either directly or through a heat exchanger to a turbine-generator.

burnable poison

A neutron absorber or poison (such as gadolinium) that, when purposely incorporated in the fuel or fuel clad- ding of a nuclear reactor, gradually "burns up" (is changed into nonabsorbing material) under neutron irradiation. This process compensates for the loss of reactivity that occurs when fuel is consumed and fission- product poisons accumulate, and keeps the overall characteristics of the reactor nearly constant.

capacity factor

The ratio of the average power load of an electric power plant to its rated capacity. Sometimes called "plant factor."

carryover

The weight-fraction of liquid water carried as droplets with the steam as it leaves the reactor. Water droplets cause erosion in the turbine and carry activated corrosion products and fission products into the turbine, potentially causing maintenance problems. Carryover to the turbine is typically kept below 0.1 wt%.

carryunder

The weight-fraction of steam bubbles entrained in the recirculating reactor water, i.e., not separated out and sent to the turbine. The entrained steam reduces the subcooling at the reactor inlet and thus increases the local and exit steam qualities of the fuel assemblies and core average voids. It may also aggravate or cause recirculation or jet pump cavitation. Carryunder is principally a drag on plant efficiency, whereas carryover is a hazard to the turbine.

cavitation

Bubble formation and collapse at a low-pressure point in a flowing stream. Bubbles will form where the local pressure falls below the vapor pressure of the liquid and will then be transported with the liquid, and collapse where the pressure is above the vapor pressure. This is a common potential problem with large pumps when operated at "off-standard" conditions.

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criticality	The state of a nuclear reactor when it is sustaining a chain reaction.
daughter	A nuclide formed by the radioactive decay of another nuclide, called in this context the parent.
decay heat	The heat produced by the decay of radioactive nuclides. Decay heat is released in a reactor following shutdown, first from fissions caused by delayed neutrons and heat capacity of core components and, over a longer period, by the radioactive decay of fission products in the reactor. This requires provision for cooling a reactor for long periods of time following shutdown.

decontamination	The removal of radioactive contaminants from surfaces or equipment, as by cleaning and washing with chemicals.
derate	Any administrative action which limits the reactor power to a value less than the nameplate rating. Derating can be used for cycle stretch-out. It may or may not involve a physical limitation in plant capability.
detergent waste filter	The detergent waste filter removes lint and other particulate matter from the radioactive decontamination drains. The filter is equipped with inlet and outlet connections in addition to vent and drain connections.
doppler effect	An increase in neutron absorption by a material as a result of an increase in its temperature. The Doppler effect of the fertile material in a reactor is an important factor for achieving safety in large thermal and fast reactors because of the decrease in reactivity with temperature.
doppler (fuel temperature coefficient)	The change in the core reactivity level for a unit change in the fuel temperature. The change results from the broadening effects of temperature on the neutron absorption resonance.
doubling time	The time for the neutron flux level to double.
driving flow	Driving flow (sometimes called recirculation flow) measurements are commonly made of the DP across the pump(s) or venturi. This is the flow through the recirculation pumps.
dryer/separator canal	The dryer and separator canal provide underwater access between the dryer-separator storage pool and the reactor well.
dryer/separator storage pool	The pool is located on the refueling floor and provides storage and servicing facilities for the dryer and separator when they have been removed from the reactor vessel.
drywell	The containment vessel enclosing the reactor and recirculation system and forming part of the primary pressure suppression system.
electrical capacity factor	The ratio of electrical energy produced in a given time interval to the electrical energy that would have been produced in that same interval if the turbine-generator were to operate continuously as its maximum rating. Note that the thermal capacity factor and the electrical capacity factor would, in general, be different and that the thermal capacity factor is the term that is more generally useful in any problem involving fuel

exposure.

elevation head — Pressure exerted by a column of fluid which is proportional to the density and height of the column. (The difference in elevation head of the boiling fuel channels and a subcooled annular region provides the pressure driving head of natural circulation reactors.)

enrichment — A process by which the relative abundances of the isotopes of a given element are altered, thus producing a form of the element that has been enriched in one particular isotope.

enthalpy — Internal energy plus mechanical equivalent of heat energy contained by a unit mass of fluid.

excess reactivity — More reactivity than that needed to achieve criticality. Excess reactivity is built into a reactor (by using extra fuel) in order to compensate for fuel burnup and the accumulation of fission-product poisons during operations.

exclusion area — An area immediately surrounding a nuclear reactor where human habitation is prohibited to ensure safety in the event of an accident.

exit quality — Quality existing at the effluent end of a fuel channel or of the entire core.

fast fission — Fission resulting from the collision of a nucleus and a high-energy neutron. Some nuclei, such as those of U-238, fission only by fast neutrons.

fertile — Capable of being transformed into a fissionable substance by capture of a neutron. Fertile material, not itself fissionable by thermal neutrons, can be converted into a fissile material by irradiation in the reactor. Common examples are U-238, Th-232 and Pu-240. When these fertile materials capture neutrons, they are partially converted into fissile Pu-239, U-233 and Pu-241, respectively.

film boiling — See "boiling"

filter/demineralizer — Removes chlorides, sulfides, oxides, etc. from radwaste effluent using Powdex resins or other suitable filtration material.

fission fragments — The two nuclei that are formed by the fission of a nucleus. Also referred to as primary fission products. They have medium atomic weights and are radioactive.

fission gas	Those fission products which exist in the gaseous state at normal temperatures and pressures.
fission, nuclear	The division of a heavy nucleus into two approximately equal parts. For the heaviest nuclei the reaction is highly exothermic, the release of energy being about 210 MeV per fission. A well-known example is the spontaneous fission of U-238. Other examples are the fissions of U-233 and Pu-239 after neutron capture.
fission poisons	Fission fragments that readily absorb neutrons; for example, Xe-135, which has an absorption cross section of 3.5 million barns for slow neutrons.
fission-product poisoning	The absorption or capture of neutrons by fission products in a reactor, thereby decreasing its reactivity.
fission products	The nuclei (fission fragments) formed by the fission of heavy elements, plus the nuclides formed by the fission fragments' radioactive decay. Some of the fission products are, or become, strong neutron absorbers.
fissionable	Capable of being fissioned by the capture of a particle, such as a neutron or photon. The most common fissionable materials are U-235, Pu-239, and U-233.
flux; neutron flux (see also "heat flux")	A measure of the intensity of neutron radiation. It is the number of neutrons passing through 1 square centimeter of a given target in 1 second. Expressed as nv, where n = the number of neutrons per cubic centimeter and v = their velocity in centimeters per second.
forced circulation	Forced circulation is induced partially by rotating and partially by jet pumps. The gross core flow and the individual fuel assembly flows are weak functions of power.
fuel assembly	A fuel assembly is a "bundle" of fuel rods held in a rigid rectangular array by tie-plates at the top and bottom, supported at intermediate levels with spacers, and enclosed by a fuel channel.
fuel bundle sampler	The device for obtaining water or gas samples from a fuel bundle in the shutdown reactor or in the storage pool. (Also called the "sipper.")
fuel storage pool	A pool that provides storage and servicing facilities for activated fuel

	elements.
fuel time-constant	The specific heat of the UO-2, combined with its thermal conductivity, yields a fuel temperature time-constant of a few seconds. The time-constant is the time required for the temperature to change by a fraction 1/e of the steady-state temperature differences associated with two flux levels if the flux level change is made as a step. This few-second fuel time constant is a very important characteristic of all UO_2 fueled reactors.
gamma radiation (γ)	High-energy, short-wavelength, electromagnetic radiation.Gamma radiation frequently accompanies alpha and beta emissions and always accompanies fission. Gamma rays are very penetrating and are best stopped or shielded against by dense materials, such as lead or depleted uranium. The rays are similar to X rays, but usually are more energetic, and are nuclear in origin.
half-life	The time in which half the atoms of a particular radioactive substance disintegrate to another nuclear form. Each radionuclide has a unique half-life. Measured half-lives vary from millionths of a second to billions of years.
heat flux	Rate of heat flow across a boundary (usually cladding surface), expressed in Btu/hr-ft or watts/cm^2.
hertz (Hz)	Unit of frequency, equal to one cycle per second.
hot	A slang term meaning highly radioactive.
hot spot	A surface area of higher-than-average radioactivity. Also a part of a fuel element surface that has one of the highest heat fluxes in the core.
important to safety	Those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Encompasses the broad class of plant features that contribute in an important way to the safe operation and protection of the public in all phases and aspects of facility operations. Includes Safety Grade or Safety-Related as a subset.
instrument channel	An arrangement of sensor and associated components used to evaluate plant variables and produce discrete outputs used in logics. A channel terminates and loses its identity where individual channel outputs are combined in logics.

integrated neutron flux	Flux multiplied by time; usually expressed as nvt, where n = the number of neutrons per cubic centimeter, v = their velocity in centimeters per second, and t = time in seconds.
interlock	A device usually electrical and/or mechanical, to prevent activation of a control until a preliminary condition has been met, or prevent hazardous operations. Its purpose usually is safety.
ion exchange	A chemical process involving the reversible interchange of various ions between a solution and solid material, usually a plastic or resin. The process is used to separate and purify chemicals, such as fission products and rare earths, in solutions.
ionization	The process of creating ions by adding or subtracting one or more electrons to or from atoms or molecules. High temperatures, electrical discharges, or nuclear radiation can cause ionization.
irradiation	Exposure to radiation, as in a nuclear reactor.
isolated condition	Normal isolation of the reactor from the main condenser, including the closure of the main steam line isolation valves.
jet pump flow	The same (except for the relatively insignificant control rod drive flow) as total core flow. The driving flow, or recirculation flow, mixes with suction flow in the throat and diffuser of the jet pump before entering the core.
leakage, neutron	The loss of neutrons from a reactor core by outward diffusion. When there is a reflector, leakage refers to net loss of neutrons that leave the core and are not reflected back into it. Leakage lowers the neutron level in a reactor.
leakage flow	Coolant flow that is diverted to other regions of the reactor outside the fuel channels to remove heat from control rods, sources and fission chambers. About 10% of the total core flow is leakage flow, removing about 3% of the heat generated in the core.
limiting conditions for operation (LCO)	Specify the minimum acceptable levels of system performance necessary to ensure safe startup and operation of the facility. When the conditions are met the plant can be operated safely and abnormal situations can be safely controlled.
limiting safety system	Instrumentation settings which initiate automatic protective action

setting (LSSS)	at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings.
linear heat generation rate (LHGR)	The heat generation system rate per unit length of a fuel rod. Common units are kW/ft.
load following	A term describing a power plant whose power is raised and lowered to meet the day-to-day demands of its electrical grid. A load-following plants the opposite of a base-loaded plant.
local peaking factor	Ratio of the maximum-to-average fuel rod power within a fuel assembly.
local power	The power generation in an arbitrary unit of volume, usually a small length of a fuel assembly called a node. It is the integral of the heat flux over the heat transfer area in the unit of volume or length, plus an increment for the heat deposited in the water by thermalization of neutrons and absorption of gamma energy where applicable.
logic	That array of components which combines individual bistable output signals to produce decision outputs.
maximum average planar linear heat generation rate (MAPLHGR)	The maximum in-core value of average planar linear Heat generation rate.
megawatt-day per ton	A unit for expressing the burnup of fuel in a reactor; specifically, the number of megawatt-days of heat per metric ton.
minimum critical power ratio (MCPR)	The smallest critical power ratio existing anywhere in the core. This expression is used in place of such terms as "minimum burnout ratio" and "minimum burnout margin." The control room problem is to determine the magnitude, and the location in the core, of the MCPR. Nomograms, worksheets, and procedures are provided so that the MCPR value can be determined for the operating condition that exists. An on-line computer may be used to determine this important value. Ordinarily, it is sufficient to determine the MCPR value following any change in operating conditions, or at intervals of once per shift of base load operation.
moderator	A material used in a reactor to slow down high-velocity neutrons and increase the likelihood of further fission. Moderators commonly include

ordinary water, heavy water and graphite. Liquid moderators can also serve as the coolant. Neutrons lose energy by scattering collisions with nuclei of the moderator. A good moderator has high scattering cross section and low atomic weight.

moderator temperature coefficient
The change in the core reactivity level per a unit change in the moderator temperature. The moderator temperature coefficient of reactivity is the composite of three principal effects. These are: (1) the temperature effect in k_∞, (2) the temperature effect on core neutron leakage, and (3) the temperature effect on the control rod system worth. The latter is a large negative effect. The composite coefficient becomes less negative with fuel depletion, reaching the least negative value at the end of each fuel cycle.

moderator void coefficient
The change in the core reactivity level per a unit change in the moderator void content. The moderator void coefficient of reactivity is a composite of the same three effects as in the temperature coefficient, but refers only to in-channel changes in moderator density, i.e., voids. As in the temperature coefficient, the void coefficient becomes less negative with fuel depletion.

multiplication factor
The effective multiplication constant (K_{eff}) is the ratio of the number of neutrons present in a reactor at a given time to the number present one finite neutron lifetime earlier. The "excess reactivity" is K_{eff}-1 which can be likened to the "interest rate per neutron lifetime," and excess of 1% means that the neutron population (capital) will increase by a factor of 1.01 in a one neutron lifetime. A reactor is said to be subcritical if $K_{eff}<1$, critical if $K_{eff}=1$, and supercritical if $K_{eff}>1$.

natural circulation
The coolant (usually water) in a reactor is circulated without pumping; that is, by natural convection resulting from the different densities of relative cold and heated portions.

natural uranium
Uranium as found in nature, containing 0.7% U-235, 99.3% U-238, and a trace of U-234.

net positive suction head (NPSH)
The pressure head, in feet of fluid, acting to suppress cavitation at the elevation of interest. It is the excess of the static head over the saturation pressure corresponding to the fluid temperature. The NPSH can be increased by greater submergence of the pump below the fluid surface and/or by greater subcooling of the suction fluid.

neutron flux
See "flux."

neutron leakage	See "leakage."
noble gases	Radioactive (nonreactive) elements, such as argon, xenon, and krypton released in the fission process.
nuclear fission	See "fission, Nuclear."
nucleate boiling	See "boiling."
nv	Neutron flux, units of neutrons per second per square centimeter.
offgas	The accumulation of air through in-leakage around the BWR turbine, the fission gases present in the steam and the hydrogen and oxygen from disassociation of water; and exhausted through the steam jet air ejectors or mechanical vacuum pumps.
passive failure	A breech of a fluid pressure boundary or blockage of a process flow path.
peaking factor, total	The ratio of the maximum fuel rod surface heat flux in any assembly to the core average fuel rod surface heat flux.
plant factor	See "capacity factor."
period	The time required for one cycle of a regularly repeated series of events. In a nuclear reactor, it is the time required for the power level to change by the factor of 2.718, which is known as e (the base of natural logarithms).
poison	Any material of high absorption cross section that absorbs neutrons unproductively and hence removes them from the fission chain reaction in a reactor and decreases its reactivity. It may be in the form of poison curtains or an accumulation of material resulting from operation, especially fission products of high-neutron-absorption cross section.
power	Power is the rate of heat production, transfer or flow.
	Fission power or (neutron power) refers to the rate of the basic fission process, and this responds essentially instantaneously to a neutron flux change. Fission power can be determined only by calibrated nuclear instrumentation during a fast transient.
	Heat flux power refers to the rate of transfer of the heat from the fuel to

the coolant. In any neutron flux transient, the rate of heat transfer lags behind the fission power generation rate because of the few second time constant of the UO2 fuel. Since heat flux cannot be measured directly, transient values must be calculated from other measurements.

<u>Reactor power</u> is determined by a coolant heat-balance under steady-state conditions by measuring pressure, or temperatures, and coolant flows to determine enthalpys and heat removal and addition rates. The calibration of primary nuclear instrumentation and the basic plant heat production records depend directly on the heat-balance data.

power density	The rate of heat generated per unit volume of a reactor core.
quality, steam	The percentage of weight of water that is in the vapor phase. In fluid flow, steam quality refers to the weight-fraction of steam in the two-phase flowing mixture of steam and liquid in the channel passing the point of interest. Thus, in a fuel assembly the quality increases from zero (below the elevation where boiling begins) to the maximum or exit value at the top of the assembly. Steam quality is the important property when considering the thermal properties of the fluid.
radial peaking factor (RPF)	The ratio of the fuel assembly power or heat flux in a particular assembly to the power or heat flux of the core average fuel assembly.
radiation	The emission and propagation of energy through matter or space by means of electromagnetic disturbances that display both wave like and particle like behavior. The term radiation; such radiation is commonly classified according to frequency, as Hertzian, infrared, visible (light), ultraviolet, X-ray, and gamma-ray. By extension, corpuscular emissions, such as alpha and beta radiation, or rays of mixed or unknown type, are classified as cosmic radiation. Nuclear radiation is that emitted from atomic nuclei in various nuclear reactions, including alpha, beta, and gamma radiation and neutrons.
reactivity	A measure of the departure of a nuclear reactor from criticality. It is about equal to the effective multiplication factor minus one and is thus precisely zero at criticality. If there is excess reactivity (positive reactivity), the reactor is supercritical and its power will rise. Negative reactivity (subcriticality) will result in a decreasing power level.
recirculation flow	See "driving flow."
refueling	The removal and addition of fuel assemblies to the core; however, the term is often extended to include any and all additions, rearrangements,

	or removals of core components which affect reactivity.
refueling platform	The platform that moves over the reactor pool and the fuel storage pool to carry operators, refueling tools, and fuel.
refueling outage	Includes all of the planned operations associated with a normal refueling except those tests in which the reactor is taken out of and returned to the shutdown (more than one rod subcritical) condition. The following operations are included in refueling: planned physical movement of core components (fuel, control rods, etc.), refueling test operations (except criticality and shutdown margin tests), and planned maintenance.
roentgen equivalent man (rem)	A measure of the dose of any ionizing radiation to body tissues in terms of its estimated biological effect. A dose of 1 rad of X, gamma or ß radiation is equivalent to 1 rem.
safety-grade	This term is not explicitly used in regulations. It is equivalent to "Safety-Related", and is a subset of "Important to Safety".
safety limits	Limits below which the reasonable maintenance of the integrity of the cladding and primary systems are assured. Operation beyond such a limit may not in itself result in serious consequences, but indicate an operational deficiency subject to regulatory review.
safety related	Those structures, systems, or components designed to remain functional for the Safe Shutdown Earthquake necessary to assure required safety functions. i.e: (1) The integrity of the reactor coolant system pressure boundary (2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guidelines in 10 CFR 100 Appendix A.
saturation	Refers to the enthalpy or temperature of a liquid at which the vapor pressure equals the local pressure. At saturation, further additions of heat cause some of the liquid to change to vapor; that is, boiling occurs.
scram	The sudden shutdown of a nuclear reactor by rapid insertion of the control rods. Emergencies or deviations from normal reactor operation cause the reactor operator or automatic control equipment to scram the reactor.
secondary containment	This is attained when the reactor building is closed and the following

integrity	conditions are met: At least one door at each access opening is closed. The Standby Gas Treatment System is operable. All reactor building ventilation system automatic isolation valves are operable or are secured in the closed position.
separator/dryer storage pool	The pool is located in the refueling floor and provides storage and servicing facilities for the separator and dryer when they are removed from the reactor vessel.
service platform	The platform placed over the reactor at the level of the vessel flange to permit operators to work on the core. Normally, this platform rests on a seal surface protector.
shutdown margin	The amount of reactivity by which the reactor is subcritical. Mathematically, $1-K_{eff}$ (for $K_{eff} < 1$). The value specified normally assumes that the strongest control rod is stuck in the fully withdrawn condition.
single failure	An occurrence which results in the loss of capability of a component to perform its intended safety function when called upon. Multiple failures resulting from a single occurrence are to be considered a single failure. Fluid process systems are considered to be designed against an assumed single failure if neither a single active nor a single passive failure results in a loss of the safety function to the nuclear unit.
sipper	See "fuel bundle sampler."
spent (depleted) fuel	Nuclear reactor fuel that has been irradiated to the extent that it can no longer effectively sustain a chain reaction.Fuel becomes spent when its fissionable isotopes have been partially consumed and fission - product poisons have accumulated in it.
stack	The chimney used to disperse the offgas from reactor operation.
storage and handling facilities, fuel	Fuel storage and handling facilities and procedures are designed to ensure that an unintentional criticality cannot occur. Any proposed change in equipment, geometry, or procedure involving either new or exposed fuel must be rigorously examined for a change in criticality risk.
subcooled boiling	See "boiling."
subcooling	The difference between the saturation enthalpy and the actual enthalpy of water in the liquid phase. Common units are Btu/lb, cal/gm, and ^{0}F.

suction flow	Suction flow is a mixture of subcooled feedwater and saturated water discarded by the steam separators and dryers above the jet pump suction inlet. This suction flow is drawn into the jet pump by the venturi effect of the driving flow.
superheating	A heating of a vapor, particularly saturated (wet) steam, to a temperature higher than the boiling point at the existing pressure. This is done in power plants to improve efficiency and to reduce condensation in the turbines.
temperature coefficient of reactivity	The change in reactor reactivity (per degree of temperature) occurring when the operating temperature changes. The coefficient is positive when an increase in temperature increases the reactivity and negative when an increase in temperature decreases reactivity. Negative temperature coefficients help to prevent power excursions. See"Doppler (Fuel Temperature) Coefficient" and "Moderator Temperature Coefficient."
thermal efficiency	The ratio of the electric power produced by a power plant to the amount of heat produced by the fuel; a measure of the efficiency with which the plant converts thermal energy to electrical energy.
thermal (slow) neutron	A neutron in thermal equilibrium with its surrounding medium. Thermal neutrons are those that have been slowed down by a moderator to an average speed of about 2200 meters per second (at room temperature) from the much higher initial speeds they had when expelled by fission. This velocity is similar to that of gas molecules at ordinary temperatures.
trip system	An arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function.
tripped (technical specifications)	The change of state of a bistable device which represents the change from a normal condition. A trip signal, which results from a trip, is generated in the channels of a trip system and produces subsequent trips and trip signals through the system as directed by the logic.
two-phase pressure drop	The DP produced by the flow of a mixture of liquid and vapor through a resistive path. The DP for a given mass flow rate is dependent on the weight fraction of the mass flow that is steam.
velocity limiter	An integral part of the control rod designed to limit the free-fall velocity

	of the control rod in water.
void coefficient	The change in reactivity resulting from a percentage change in void fraction.
void fraction	Fraction of the volume of the coolant stream (moderator) that is in the vapor phase. Void fraction is an important property when considering nuclear properties such as reactivity, moderation, etc.

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Appendix B

Piping and Instrumentation Symbols

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Appendix C

List Of Common Abbreviations

This Collection of Abbreviations is taken in part from NUREG-0544/Rev 3
"NRC Collection of Abbreviations."

LIST OF COMMON ABBREVIATIONS (FOR BOILING WATER REACTORS)

A	TERM
ac	Alternating current
A/C	Air conditioning
ACU	Analog comparator unit
ACC	Accumulator
A/D	Analog to digital
ADS	Automatic depressurization system
AHU	Air handling unit
AID	Alarm initiated display
ALF	Automatic load following
AO	Air operator
APF	Axial peaking factor
APLHGR	Average planar linear heat generation rate
APRM	Average power range monitor
ARM	Area radiation monitor
ASM	Auxiliary select module
ATWS	Anticipated transient without scram
ATWS-RPT	Anticipated transient without scram recirculation pump trip

B	
BAF	Bottom of active fuel
BOC	Beginning of cycle
BOL	Beginning of life
BOP	Balance of plant
BPV	Bypass valve
BUOT	Backup overspeed trip
BWR	Boiling water reactor

C	
CAEQ	Core average exit quality
CB	Circuit breaker
CCGC	Containment combustible gas control
CCW	Closed cooling water or Condenser circulating water
C/D	Cooldown
CIV	Combined intermediate valve
CMFLPD	Core maximum fraction of limiting power density
CMPF	Core maximum peaking factor
cpm	Counts per minute

CPR	Critical power ratio
cps	Counts per second
CR	Control rod
CRD	Control rod drive or Control rod density
CRDH	Control rod drive hydraulic
CRDM	Control rod drive mechanism
CRW	Clean radwaste
CS	Controlled shutdown
CSCS	Core standby cooling systems
CST	Condensate storage tank
CTS	Condensate transfer and storage
CV	Control valve
CWS	Circulating water system

D

DAP	Data acquisition processor
DBA	Design basis accident
dc	Direct current
DCP	Display control processor
DCPS	DC power system
DCS	Display control system (BWR/6)
DCV	Directional control valve
D/G	Diesel generator
DG	Display generator (BWR/6)
DNB	Departure from nucleate boiling
DRW	Dirty radwaste
DVM	Digital volt meter
DW	Drywell
DWEDS	Drywell equipment drain sump
DWFDS	Drywell floor drain system
DWS	Demineralized water system

E

EACPS	Essential ac power system
EBOP	Emergency bearing oil pump
ECCS	Emergency core cooling systems
ED/G	Emergency diesel generator
EECW	Emergency equipment cooling water
EGS	Electronic governing system
EHC	Electro hydraulic control
EMF	Electro motive force
ENR	Extraction nonreturn
EOC	End of cycle

EOC-RPT	End of cycle recirculation pump trip
EOF	Emergency operations facility
E/P	Electrical/pneumatic
EPR	Electric pressure regulator
ERF	Emergency response facility
ERFIS	Emergency response facility information system
ESF	Engineered safety feature
ETS	Emergency trip system
ETV	Electric trip valve

F

FB	Fuel building
FCPM	Feedwater corrosion product monitor
FCU	Fan coil unit
FCV	Flow control valve
F/D	Filter/demineralizer
FLC	Fuel loading chamber
FPCC	Fuel pool cooling and cleanup
FS	Flow switch
FSCRD	Fast scram control rod drive
FT	Flow transmitter
FWCI	Feedwater coolant injection (BWR/2, BWR/3)
FWCS	Feedwater control system
FWCV	Feedwater control valve
FWOL	Fuel warranty operating limits

G

GAF	Gain adjustment factor
GEXL	General Electric critical quality (X_c)-boiling length (LB) correlation
GM	Geiger-Mueller
gpm	Gallons per minute
GSLO	Gland seal leak off

H

HAD	Heat actuated device
HCU	Hydraulic control unit
HEPA	High efficiency particulate air
HIS	Hydrogen ignition system
HO	Hydraulic operator
HP	High pressure
HPCI	High pressure coolant injection
HPCS	High pressure core spray (BWR/5, BWR/6)
HPSP	High power setpoint

HPSU	Hydraulic power supply unit
HPU	Hydraulic power unit
H/U	Heat-up
HVAC	Heating, ventilation, and air conditioning
HX	Heat exchanger

<u>I</u>

IC	Isolation condenser (BWR/2, BWR/3)
I/O	Input/output
I/P	Current/pneumatic
IRM	Intermediate range monitor
IV	Intercept valve

<u>K</u>

kV	Kilovolts
kW	Kilowatts

<u>L</u>

LBS	Load break switch
LCR	Log count rate
LD	Leak detection
LED	Light emitting diode
LFMG	Low frequency motor generator (BWR/5, BWR/6)
LHGR	Linear heat generation rate
LI	Level indicator
LIC	Level indicating controller
LIS	Level indicating switch
LITS	Level indicating transmitting switch
LOCA	Loss of coolant accident
LOPP	Loss of preferred power
LP	Low pressure
LPAP	Low power alarm point
LPCI	Low pressure coolant injection
LPCS	Low pressure core spray (BWR/5, BWR/6)
LPF	Local peaking factor
LPRM	Local power range monitor
LPSP	Low power setpoint
LR	Liquid radwaste or Load reject
LS	Level switch
LT	Level transmitter
LTNGP	Low temperature noble gas process
LU	Logic unit
LVDT	Linear variable differential transformer

LVT	Linear velocity transformer
LWR	Light water reactor
M	
M/A	Manual/automatic
MAPLHGR	Maximum average planar linear heat generation rate
MAT	Main auxiliary transformer
MCC	Motor control center
MCPR	Minimum critical power ratio
MFLCPR	Maximum fraction of limiting critical power ratio
MG	Motor generator
MLV	Mechanical lockout valve
MO	Motor operator
MOV	Motor operated valve
MPR	Mechanical pressure regulator (BWR/2, BWR/3)
MS	Main steam
MSIV	Main steam isolation valve
MSIV-LCS	Main steam line isolation valve leakage control system
MSL	Main steam line
MSOP	Main shaft oil pump
MSR	Moisture separator reheater
MSSV	Main steam shutoff valve
MSV	Mean square voltage
MTH	Manual trip handle
MTP	Mechanical trip piston
MTPV	Mechanical trip pilot valve
MTSV	Mechanical trip solenoid valve
MTU	Master test unit
MTV	Mechanical trip valve
MWe	Megawatt electric
MWt	Megawatt thermal
N	
NAACP	Normal auxiliary ac power
NCC	Nuclenet control console (BWR/6)
NDL	Nuclear data link
NDTT	Nil ductility transition temperature
NMS	Neutron monitoring system
NPSH	Net positive suction head
NRHX	Nonregenerative heat exchanger
NSPS	Nuclear safety protection system (BWR/5, BWR/6)
NSS	Nuclear steam system
NSSS	Nuclear steam supply system

NSSSS	Nuclear steam supply shutoff system (BWR/5, BWR/6)
nv	Neutron flux
O	
OCM	Operator control module
OD	On demand
ODYN	One dimensional core transient model
OG	Offgas
OI	Optical isolator
OSC	Operational support center
OST	Overspeed trip
OSTV	Oil trip solenoid valve
OTB	Onset of transition boiling
P	
P/B	Pushbutton
PC	Process computer or Primary containment
PCI	Pellet cladding interaction
PCIOMR	Pre-conditioning interim operating management recommendation
PCIS	Primary containment isolation system
PCT	Peak cladding temperature
PCV	Pressure control valve
PGCC	Power generation control complex (BWR/6)
PI	Pressure indicator
PIS	Pressure indicating switch
PLU	Power load unbalance
PMS	Performance monitoring system (BWR/6)
PP	Power panel
ppb	Parts per billion
ppm	Parts per million
PRM	Process radiation monitor
PS	Pressure switch
PT	Pressure transmitter
R	
RACS	Rod action control system
RAT	Reserve auxiliary transformer
RAU	Remote analog unit
RBCCW	Reactor building closed cooling water
RBEDT	Reactor building equipment drain tank
RBM	Rod block monitor
RCIC	Reactor core isolation cooling
RCIS	Rod control and information system

RDA	Rod drop accident
RDM	Rod display module
RDU	Remote digital unit
RECHAR	Recomber charcoal adsorber
RFC	Recirculation flow control
RFP	Reactor feed pump
RFPT	Reactor feed pump turbine
RGDS	Rod gang drive system (BWR/6)
RHR	Residual heat removal
RHRSW	Residual heat removal service water
RHX	Regenerative heat exchanger
RIS	Rod interface system
RM	Radiation monitoring
RMCS	Reactor manual control system
RMS	Remote manual switch
RPC	Rod pattern controller
RPF	Radial peaking factor
RPIS	Rod position information system
rpm	Revolutions per minute
RPS	Reactor protection system
RPT	Recirculation pump trip
RPV	Reactor pressure vessel
RSCS	Rod sequence control system (BWR/4, BWR/5)
RSS	Remote shutdown system
RTD	Resistance temperature detector
RVDT	Rotary variable differential transformer
RWB	Rod withdraw block
RWCU	Reactor water cleanup
RWM	Rod worth minimizer
RX	Reactor

S

SACPS	Standby ac power system
SBS	Static bypass switch
SCM	Steam condensing mode
SCR	Silicon control rectifier
S/D	Shutdown
SDC	Shutdown cooling
SDM	Shutdown margin
SDIV	Scram discharge instrument volume
SDV	Scram discharge volume
SGTS	Standby gas treatment system
SIA	Service and instrument air

SIP	Standby information panel
SJAE	Steam jet air ejector
SLC	Standby liquid control
SLF	Steam line flow
SLP	Steam line pressure
SP	Suppression pool
SPC	Suppression pool cooling
SPDS	Safety parameter display system
SPE	Steam packing exhauster
SPMS	Suppression pool makeup system
SRM	Source range monitor
SRV	Safety /relief valve
SSE	Steam seal evaporator or Safe shutdown earthquake
SSR	Steam seal regulator
SSW	Station service water
SSWS	Standby service water system
S/U	Startup
SUS	Secondary unit substation
SV	Stop valve

<u>T</u>

TAF	Top of active fuel
TBCCW	Turbine building closed cooling water
TCV	Temperature control valve
TD	Time delay
TG	Turbine generator
TGOP	Turning gear oil pump
TGSS	Turbine gland sealing system
TIP	Traversing incore probe
TLO	Turbine lube oil
TPF	Total peaking factor
TSC	Technical support center

<u>U</u>

UHS	Ultimate heat sink
UPS	Uninteruptible power supply

<u>V</u>

Vac	Volts alternating current
Vdc	Volts direct current

Boiling Water Reactor

GE BWR/4 Technology

Technology Manual

Appendix D

Purpose and Objectives

Course Objectives
(R-304B)

The General Electric technology systems course is designed to provide the student with a comprehensive understanding and working knowledge of the boiling water reactor (BWR/4) commercial steam electric plant. At the end of this course each student should have achieved a basic understanding of the following:

- Nuclear theory, reactivity coefficients, and thermal limits,
- Process systems, purposes, theory of operation, normal system configuration, and safety related flowpaths and/or operations,
- Plant electrical system design and distribution,
- Process instrumentation systems including, logics, selected interlocks, limiters, control and protection functions,
- PRA insights into assessing a change to the level of plant safety/risk as a result of system or component problems or failures.

BWR INTRODUCTION

(1.0)

PURPOSE

To economically generate electrical power through the use of the direct cycle Boiling Water Reactor (BWR) System design.

This design includes the nuclear fuel and internal structures within the Reactor Pressure Vessel, systems associated with a basic steam cycle, normal auxiliary systems to accommodate the operation requirements of the plant, and the necessary instrumentation and controls to accommodate operator control of the plant.

LESSON OBJECTIVES

1. Explain the basic steam cycle as applied to BWR systems.
2. State which BWR control systems are used for the following important functions:
    a. Control of reactor power
    b. Control of reactor pressure (normal situations)
    c. Control of reactor water level
3. State the type of containment package which is provided and explain the following terms:
    a. Drywell
    b. Suppression pool
    c. Containment
4. List the four Emergency Core Cooling Systems and state which are high pressure systems and which are low pressure systems.

REACTOR PHYSICS

(1.7)

LESSON OBJECTIVES

1. List the three BWR coefficients of reactivity and state how and why they change with core life and temperature.
2. Define the following reactor physics terms:
   a. Reactivity
   b. Keff
   c. Reactor period
   d. Shutdown margin
3. Describe the response of fission product poisons xenon and samarium to changes in reactor power.
4. Describe the initial plant response (via the reactivity coefficients) to changes in significant plant parameters.

THERMAL LIMITS

(1.8)

PURPOSE

To minimize the radiological release from the plant by ensuring that fuel cladding integrity is maintained.

LESSON OBJECTIVES

1. List the three basic BWR thermal limits and explain the purpose of each.
2. Define fuel damage and explain the mechanisms which could cause fuel damage.
3. Define the following terms used in the study of BWR thermal limits:
    a. Linear heat generation rate (LHGR)
    b. Average planar linear heat generation rate (APLHGR)
    c. Critical power
    d. Critical power ratio (CPR)
    e. Safety limit MCPR
    f. MCPR operating limit

REACTOR VESSEL SYSTEM

(2.1)

PURPOSES

1. To house the reactor core.
2. To support and align the fuel and control rods
3. To provide a flow path for the circulation of coolant past the fuel.
4. To remove moisture from the steam exiting the reactor vessel.
5. To provide an internal, refloodable volume to assure core cooling capability following a Loss of Coolant Accident (LOCA).
6. To serve as part of the reactor coolant boundary.
7. To limit downward control rod motion following a postulated failure of a control rod drive housing.

LESSON OBJECTIVES

1. State the system's purposes.
2. Place selected major reactor vessel components in flowpath order, explain the purpose of each, and indicate whether flow through them is water and/or steam.
3. Describe the internal components and their arrangement and how that provides for the 2/3 core coverage capability following a LOCA.
4. Explain how this system interfaces with the following systems:
    a. Control Rod Drive System
    b. Main Steam System
    c. Condensate and Feedwater System
    d. Residual Heat Removal System
    e. Core Spray System.
    f. Neutron Monitoring Systems
    g. Reactor Water Cleanup System
    h. Standby Liquid Control System

FUEL

(2.2)

PURPOSE

To generate energy from a nuclear fission reaction to provide heat for steam generation.

LESSON OBJECTIVES

1. State the system's purpose.
2. Describe the physical arrangement of:
   a. The fuel assemblies
   b. A fuel rod and its internal components
3. State the purpose of the major system components.
4. Explain the characteristics of the fuel with regard to:
   a. Fuel pellet composition
   b. Enrichment variations
   c. Burnable poisons
5. Discuss the function of core bypass flow and its sources.

CONTROL RODS

(2.2)

PURPOSES

1. To control reactor power level.
2. To control axial and radial power (neutron flux) distribution to optimize core performance.
3. To provide adequate excess negative reactivity to shut down the reactor from any normal operating or accident condition at any time during core life.

LESSON OBJECTIVES

1. State the system's purposes.
2. Describe the physical arrangement of the control rods in the reactor vessel.
3. State the purpose of the major control rod components.
4. Describe the nuclear characteristics of the control rods.
5. Explain why bottom entry control rods are used.

CONTROL ROD DRIVE SYSTEM

(2.3)

PURPOSES

1. To make gross changes in core reactivity by manually positioning the control rods in response to the Reactor Manual Control System (RMCS) signals.
2. To rapidly insert all control rods to shut down the reactor in response to Reactor Protection System (RPS) signals.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Place major system components in flow path order and explain the purpose of each.
   a. CRD pumps
   b. Recirculation Pump Seal Water Line
   c. Charging Water Line and Accumulators
   d. Flow Control Valves and Flow Sensor
   e. Drive Water Line and Directional Control Valves
   f. Cooling Water Line and Drive/Cooling Water Pressure Control Station
   g. Stabilizing Valves
4. Describe the basic flow paths within the system.
5. Discuss the coupling of the CRDM to the control rod.
6. Describe the rod position information system sensor arrangement.
7. Explain how this system interfaces with the following systems or components:
   a. Condensate and Feedwater System
   b. Service and Instrument Air System
   c. Control rods
   d. Reactor Manual Control System
   e. Reactor Protection System
   f. Recirculation System
   g. Reactor Water Cleanup System
   h. Standby AC System

RECIRCULATION SYSTEM

(2.4)

PURPOSE

To provide forced circulation of water through the reactor core, permitting higher reactor power than with natural circulation.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purpose.
3. Place major system components in flow path order and explain the purpose of each.
    a. RHR penetrations
    b. RWCU penetrations
    c. Temperature elements
    d. Flow elements
    e. Suction and discharge valves
    f. Pumps
4. Explain how the recirc pump seal assembly indications are affected on seal failures.
5. Explain how this system interfaces with the following systems:
    a. Recirculation Flow Control System
    b. Residual Heat Removal System
    c. Reactor Water Cleanup System
    d. Reactor Vessel System
    e. Control Rod Drive System
    f. Average Power Range Monitoring System
    g. Rod Block Monitoring System
    h. Feedwater Control System
    i. Reactor Building Closed Loop Cooling Water System
    j. Liquid Radwaste System

MAIN STEAM SYSTEM

(2.5)

PURPOSES

1. To direct steam from the reactor vessel to the main turbine and other steam loads.
2. To provide overpressure protection for the Reactor Coolant System.
3. To direct steam to certain safety related equipment.

LESSON OBJECTIVES

1. State the system's purposes.
2. Place major system components in flow path order and explain the purpose of each.
   a. Safety / Relief Valves
   b. Main Steam Line Flow Restrictors
   c. Main Steam Line Isolation Valves
   d. Equalizing Header
   e. Turbine Bypass Valves
   f. Main Turbine
   g. Extraction Steam System
   h. Moisture Separator Reheater
3. Explain the different modes of safety/relief valve operation.
4. List the signals which automatically close the main steam isolation valves and explain the reason for the isolation.
5. Explain how this system interfaces with the following systems:
   a. Reactor Vessel System
   b. Reactor Core Isolation Cooling System
   c. Offgas System
   d. Electro Hydraulic Control System
   e. Nuclear Steam Supply Shutoff System
   f. Residual Heat Removal System
   g. Automatic Depressurization System
   h. Condensate and Feedwater System
   i. Reactor Protection System
   j. Feedwater Control System

CONDENSATE AND FEEDWATER SYSTEM

(2.6)

PURPOSES

1. To condense steam, to purify, preheat, and pump water from the main condenser to the reactor vessel.
2. To provide a means for the Reactor Water Cleanup (RWCU) System and certain safety related systems (HPCI and RCIC) to discharge water to the reactor vessel.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Place major system components in flow path order and explain the purpose of each.
    a. Main Condenser
    b. Condensate Pumps
    c. Condensate Demineralizers
    d. Condensate Booster Pumps
    e. Low Pressure Heaters
    f. Feedwater Pumps
    g. High Pressure Heaters
    h. Reactor Vessel
4. Explain this system interfaces with the following systems:
    a. Reactor Vessel System
    b. Reactor Core Isolation Cooling System
    c. High Pressure Coolant Injection System
    d. Reactor Water Cleanup System
    e. Feedwater Control System
    f. Main Steam System
    g. Offgas System

REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

(2.7)

PURPOSE

To provide makeup water to the reactor vessel for core cooling when:

1. The main steam lines are isolated.
2. The Condensate and Feedwater System is not available.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Place major system components in flow path order and explain the purpose of each.
    a. Suction valve
    b. RCIC pump
    c. Minimum flow valve
    d. Injection valve
    e. Air operated check valve
4. State the system's initiation, isolation, and turbine trip signals.
5. Explain the system's response to an automatic initiation, isolation, and turbine trip.
6. Explain the interfaces this system has with the following systems or components:
    a. Main Steam System
    b. Condensate and Feedwater System
    c. Suppression pool
    d. Condensate storage tank
    e. Nuclear Steam Supply Shutoff System

REACTOR WATER CLEANUP (RWCU) SYSTEM

(2.8)

PURPOSES

1. To maintain reactor water quality by filtration and ion exchange.
2. To provide a path for removal of reactor coolant from the vessel.
3. To aid circulation in the reactor vessel bottom head to minimize thermal stratification.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. State why reactor water purity is maintained and how it is monitored.
4. Discuss the system's isolation and pump trip signals.
5. Explain how this system interfaces with the following systems:
    a. Reactor Recirculation System
    b. Reactor Vessel System
    c. Condensate and Feedwater System
    d. Reactor Building Closed Loop Cooling Water System

REACTOR VESSEL INSTRUMENTATION SYSTEM

(3.1)

PURPOSE

To provide sufficient information concerning reactor vessel water level, reactor vessel pressure, reactor vessel temperature, core flow rate, and below and above core plate pressure to:

1. Allow for proper plant operation.
2. Provide initiation signals for safety systems.

LESSON OBJECTIVES

1. State the system's purposes.
2. Describe the ranges of reactor vessel water level instrumentation and the uses for each range.
3. State the other monitored parameters.
4. State the reason for each signal listed in tables 3.1-1 and 3.1-2.

ELECTRO HYDRAULIC CONTROL (EHC) SYSTEM

(3.2)

PURPOSES

1. To provide normal reactor pressure control by controlling steam flow consistent with the amount being generated in the reactor.
2. To control reactor pressure during startup, heatup, and cooldown evolution.
3. To control the speed and electrical load on the turbine generator.
4. To provide protection for the main turbine.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Explain the necessity of reactor pressure control to BWR operation.
4. Explain how reactor pressure is controlled during all modes of plant operation.
5. Explain system operation by using Figure 3.2-1.

FEEDWATER CONTROL SYSTEM (FWCS)

(3.3)

PURPOSE

To control the rate of feedwater flow to the reactor vessel to maintain the proper vessel water level.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purpose.
3. List the parameters used by the system.
4. Explain the uses of the total steam flow, total feedwater flow, and reactor vessel water level signals.
5. Discuss the components controlled by the system.
6. Discuss the modes of control used by the system.
7. Explain how this system interfaces with the following systems:
   a. Reactor Vessel Instrumentation System
   b. Main Steam System
   c. Condensate and Feedwater System
   d. Reactor Protection System
   e. Recirculation Flow Control System.
   f. Rod Worth Minimizer

PRIMARY CONTAINMENT SYSTEM

(4.1)

PURPOSES

1. To condense steam and contain fission products released from a loss of coolant accident (LOCA) so that offsite radiation doses specified in 10 CFR 100 are not exceeded.
2. To provide a heat sink for certain safety related equipment.
3. To provide a source of water for Emergency Core Cooling Systems and the Reactor Core Isolation Cooling System.

PRIMARY CONTAINMENT SYSTEM

(4.1)

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Explain the general arrangement and purpose of the following major system components:
   a. Drywell
   b. Vertical downcomers (vents)
   c. Suppression chamber
   d. Suppression pool
   e. Suppression chamber to drywell vacuum breakers
   f. Primary Containment Purge System
4. Explain the multibarrier, pressure suppression concept as applied to the Mark II Containment design.
5. Describe what constitutes the primary containment fission product barrier.
6. Explain the primary containment response to a major loss of coolant accident.
7. Explain how POST-LOCA hydrogen gas concentration is controlled.
8. Define primary containment integrity.
9. Describe the interfaces between this system and the following:
   a. Main Steam System
   b. Reactor Core Isolation Cooling System
   c. Secondary Containment System
   d. Reactor Building Standby Ventilation System
   e. Nuclear Steam Supply Shutoff System
   f. High Pressure Coolant Injection System
   g. Core Spray System
   h. Residual Heat Removal System

SECONDARY CONTAINMENT SYSTEM

(4.2)

PURPOSES

1. To minimize the ground level release of radioactive material following an accident.
2. To provide primary containment when the Primary Containment System is open.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Explain the general arrangement and purpose of the following major system components:
    a. Normal Ventilation System Supply and Exhaust Isolation Valves
    b. Normal Ventilation Supply Fans Discharge Dampers
    c. Area Unit Coolers
4. Explain what constitutes the secondary containment volume.
5. Define secondary containment integrity.
6. Explain the interfaces between this system and the following:
    a. Primary Containment System
    b. Reactor Building Standby Ventilation System
    c. Nuclear Steam Supply Shutoff System

REACTOR BUILDING STANDBY VENTILATION SYSTEM

(4.3)

PURPOSES

1. To process exhaust air from the secondary containment boundary under accident conditions in order to limit radiation dose rates to less than the 10 CFR 100 guidelines.
2. To maintain a negative pressure in the secondary containment upon loss of the normal ventilation system.
3. To perform secondary containment leak tests.
4. To purge the primary containment.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Explain the general arrangement and purpose of the following major system components:
    a. Reactor Bldg. Exhaust Fans
    b. Mixing plenum
    c. Cooling coils
    d. Filter Train Booster Fans
    e. Electric heater
    f. HEPA filters
    g. Charcoal filter
    h. Flow control dampers
4. State the system initiation signals and the reason for each.
5. Describe the interfaces between this system and the following:
    a. Primary Containment System
    b. Secondary Containment System
    c. Process & Area Radiation Monitoring System
    d. High Pressure Coolant Injection System

NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM

(4.4)

PURPOSE

To isolate the primary and secondary containments during accident conditions in order to prevent release of radioactive materials to the environment in excess 10 CFR 100 limits.

LESSON OBJECTIVES

1. Explain the system's purpose.
2. Explain the logic arrangement within the Nuclear Steam Supply Shutoff System (NSSSS).
3. Explain the concept of "Group" isolations.
4. List the parameters which initiate Group I isolations, conditions under which the isolations are or can be bypassed, and the reason for having each isolation.
5. Explain how isolation can be manually initiated.
6. Explain the interfaces this system has with other plant systems.
   a. Main steam system
   b. Reactor water cleanup system
   c. Recirculation system
   d. Residual heat removal system
   e. Liquid radwaste system
   f. Reactor protection system
   g. Primary containment system
   h. Reactor core isolation cooling system
   i. High pressure coolant injection system

NEUTRON MONITORING SYSTEM (NMS)

(5.0)

PURPOSES

1. To provide neutron flux level monitoring of the reactor from shutdown conditions to power operations.
2. To detect conditions in the core that threaten the integrity of the fuel cladding resulting from excessive power generation.
3. To provide signals to the Reactor Protection System and the Reactor Manual Comtrol System.

LESSON OBJECTIVES

1. State the system's purposes.
2. List the neutron monitoring systems and:
   a. State the operational condition (shutdown, startup, heatup, or power operation) during which each system is used.
   b. Describe the arrangement of the neutron monitoring systems' detectors within the reactor core.
   c. Explain the method used by the neutron monitoring systems' detectors to monitor neutron flux, and the measures employed to prolong useful detector life (information in later sections).
   d. Explain the interfaces among the systems used for power range operation.

SOURCE RANGE MONITORING (SRM) SYSTEM

(5.1)

PURPOSE

To monitor neutron flux for display and initiation of rod blocks from shut down conditions to overlap with the intermediate range.

LESSON OBJECTIVES

1. State the system's purpose.
2. Describe the indication provided by the system.
3. List and explain each rod block imposed by this system.
4. Explain how this system interfaces with the following plant systems:
    a. Reactor Manual Control System
    b. Reactor Protection System

INTERMEDIATE RANGE MONITORING (IRM) SYSTEM

(5.2)

PURPOSES

1. To provide neutron flux information from the upper portion of the source range to the lower portion of the power range.
2. To provide trip signals to preserve the integrity of the fuel cladding.

LESSON OBJECTIVES

1. State the systems's purposes.
2. Explain how the method of ranging provides protection against rapid power increases.
3. List the protective trips generated by this system, the action caused by the trips, and the reason for the trips.
4. Explain the interfaces this system has with the following plant systems:
   a. Reactor Protection System
   b. Reactor Manual Control System

LOCAL POWER RANGE MONITORING (LPRM) SYSTEM

(5.3)

PURPOSE

To provide signals proportional to the local neutron flux at various radial and axial incore locations to the

1. Average Power Range Monitor System
2. Process Computer System

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain why calibration of the LPRMs is necessary and how it is accomplished.
3. Explain the interfaces this system has with the following plant systems:
    a. Average Power Range Monitoring System
    b. Traversing Incore Probe System
    c. Process Computer System

AVERAGE POWER RANGE MONITORING (APRM) SYSTEM

(5.4)

PURPOSES

1. To monitor the core average thermal power.
2. To provide trip signals to preserve the integrity of the fuel cladding.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. List the protective trips generated by this system, the action caused by the trip, and the reason for the trips.
4. Describe the assignment of local power range monitor detectors assigned to the average power range monitors.
5. Explain the interfaces this system has with the following plant systems:
    a. Reactor Manual Control System
    b. Recirculation Flow Control System
    c. Local Power Range Monitoring System
    d. Reactor Protection System
    e. Rod Block Monitoring System

ROD BLOCK MONITORING (RBM) SYSTEM

(5.5)

PURPOSES

1. To monitor average power around a selected control rod
2. To limit control rod movement to prevent local fuel damage.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Describe the bypasses provided for the system; including parameters, and the reasons bypassing is allowed.
4. Explain the interfaces this system has with the following plant system:
    a. Local Power Range Monitoring System
    b. Recirculation System
    c. Reactor Manual Control System
    d. Average Power Range Monitoring System

TRAVERSING INCORE PROBE (TIP) SYSTEM

(5.6)

PURPOSE

To provide a means of obtaining core power distribution.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purposes.
3. Explain the interfaces this system has with the following plant systems:
   a. Local Power Range Monitoring System
   b. Process Computer System
   c. Nuclear Steam Supply Shutoff System

PROCESS COMPUTER SYSTEM

(6.1)

PURPOSES

1. To provide on-line monitoring of significant plant process variables.
2. To scan the inputs and issue appropriate alarms and messages if analog limits are exceeded or digital trip signals are received.
3. To provide the operator with essential plant performance information through logs, trends, summaries, and data arrays.
4. Contributes significantly to the maintenance of optimum core power distribution, economical utilization of nuclear fuel, and overall plant operating efficiency.

LESSON OBJECTIVES

1. State the system's purposes.
2. State how the system accomplices it's purposes.
3. Explain how the process computer can be used to determine plant performance.

EMERGENCY RESPONSE INFORMATION SYSTEM

(6.4)

PURPOSE

To provide a concise display of critical plant variables to aid control room personnel during abnormal and emergency conditions in determining the safety status of the plant and mitigating the effects of the transients to avoid a degraded core.

LESSON OBJECTIVES

1. State the purpose of the Emergency Response Information System.
2. List and explain the types of displays provided by ERIS.

REACTOR MANUAL CONTROL SYSTEM

(7.1)

PURPOSES

1. To provide control signals to control drive system for normal rod movement.
2. To prevent control rod movement during potentially unsafe conditions.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain the uses of the Rod Motion Control Pushbuttons.
3. Explain how control rod motion is achieved.
   a. Insert
   b. Withdraw
   c. Continuous withdraw
   d. Continuous insert
4. List and describe the two types of rod blocks
5. Explain how control rod position is measured.
6. Explain how this system interfaces with the following systems:
   a. Control Rod Drive System
   b. Recirculation System
   c. Neutron Monitoring System
   d. Rod Worth Minimizer

RECIRCULATION FLOW CONTROL (RFC) SYSTEM

(7.2)

PURPOSE

To control the rate of recirculation system flow, allowing control of reactor power over a limited range.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purpose.
3. Explain how changing the core flow rate can change reactor power.
4. State the purpose of the major system components.
   a. Master controller
   b. Dual limiter
   c. M/A transfer station
   d. Start signal generator
   e. Minimum speed limiter
   f. Operational limiter
   g. Function generator
   h. Scoop tube
   i. Exciter
   j. Drive motor
   k. Fluid coupler
   l. Generator
   m. Recirculation pump trip breakers
5. Given a power to flow map, Core Flow and Reactor Power, state if operations are within the allowed operating region.
6. Explain how this system interfaces with the following systems:
   a. Recirculation System b. Feedwater Control System

REACTOR PROTECTION SYSTEM (RPS)

(7.3)

PURPOSES

To initiate a reactor scram to:

1. Preserve the integrity of the fuel cladding.
2. Preserve the integrity of the reactor coolant system.
3. Minimize the energy which must be absorbed following a Loss of Coolant Accident (LOCA).

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain the logic used by the system to generate a trip signal.
3. Explain how a trip signal causes control rod insertion, and what ensures a scram is completed once it is initiated.
4. Explain the fail-safe features of the system.
5. Given a scram signal, state the reason for each scram signal, the conditions which may bypass it, and the reason each bypass is allowed.
6. Explain the purpose of the reactor mode switch.
7. Explain how this system interfaces with the following systems:
   a. Control Rod Drive System
   b. Neutron Monitoring System
   c. Instrument Air System

STANDBY LIQUID CONTROL (SLC) SYSTEM

(7.4)

PURPOSE

To shut down the reactor by chemical poisoning from a full power condition, independent of any control rod motion, and to maintain it in a subcritical condition during plant cooldown.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purpose.
3. Explain why the system lines are heat traced.
4. List the conditions that require manual initiation of the Standby Liquid Control System.
5. List the indications that can be used to verify initiation of the Standby Liquid Control System.
6. List the positive reactivity sources considered in the calculation of the minimum average boron concentration that is required to achieve the desired Shutdown Margin of 0.05% delta K/K.
7. Explain the purpose of the major system components:
    a. Storage tank
    b. Test tank
    c. Pumps
    d. Explosive valves
    e. Accumulators

ROD WORTH MINIMIZER

(7.5)

PURPOSE

To serve as a backup to procedural controls and to limit rod worth during low power operation.

LESSON OBJECTIVES

1. State the system's purpose.
2. List the rod blocks imposed by the RWM.
3. Define black/white control rod pattern.
4. Define the following terms.
   a. Low Power Alarm Point
   b. Transition Zone
   c. Low Power Set Point
5. Explain how this system interfaces with the following systems:
   a. Main Steam System
   b. Reactor Manual Control System
   c. Process Computer

OFFGAS SYSTEM

(8.1)

PURPOSES

1. To establish and maintain a vacuum in the main condenser to improve turbine efficiency by removing non condensible gases from the main condenser
2. To process non condensible gases in order to limit the release of radioactive gases to as low as is reasonably achievable(ALARA).

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purpose.
3. Describe the sources of noncondensible gases.
4. Place major system components in flow path order and explain the purpose of each.
   a. CAR Pumps
   b. SJAE and coolers
   c. Booster SJAE's
   d. Preheater
   c. Recombiner
   d. Desuperheater condenser and drain cooler
   e. Cooler condenser
   f. Sacrificial decay beds
   g. Cyclic dryer units
   h. Charcoal adsorber tanks
   i. After filters
   j. Discharge isolation valve
   k. Station ventilation booster fans
5. Explain how this system interfaces with the following systems or components:
   a. Main Condenser
   b. Main Steam System
   c. Condensate and Feedwater System

LIQUID RADWASTE SYSTEM

(8.2)

PURPOSES

1. To collect, process, and return radioactive liquid waste to the plant for reuse.
2. To allow for batch discharge of liquid radwaste to the environment in such a manner that 10 CFR 20 radionuclide concentration standards are not exceeded and dose commitments of 40 CFR 142 and effluent technical specifications are not exceeded.

LESSON OBJECTIVES

1. State the system's purposes.
2. List the four (4) classifications of liquid radwaste.
3. Place major system components in flow path order and explain the purpose of each for the
    a. Waste Collector System.
    b. Floor Drain Collector System.
    c. Chemical Waste System.
    d. Detergent Waste System.
4. Explain the batch release concept as applied to liquid radwaste.

SOLID RADWASTE SYSTEM

(8.3)

PURPOSES

1. To collect, process, store, package, and prepare for shipment solid radwaste material produced through plant operations.
2. To comply with 10 CFR 71, 10 CFR 61, and U.S. Department of Transportation (DOT) regulations appropriate to the mode of transportation in 49 CFR 170 through 189.

LESSON OBJECTIVES

1. State the system's purposes.
2. List the three (3) classifications of solid radwaste.
3. Place major system components in flow path order and explain the purpose of each for the Wet Solid Waste System.

RADIATION MONITORING SYSTEM(8.4)

PURPOSES

LESSON OBJECTIVES

ELECTRICAL SYSTEMS

(9.1 - 9.4)

PURPOSES OF THE NORMAL AC POWER SYSTEM

1. To provide adequate power to unit auxiliary loads needed for normal operation of the plant.
2. To deliver two physically independent offsite power supplies from the utility transmission network to the Emergency AC Power System switchgear.

PURPOSE OF EMERGENCY AC POWER SYSTEM

To provide a reliable source of ac power to all loads which are required for safe shutdown of the plant.

PURPOSES OF THE 120 VAC POWER SYSTEM

1. To provide 120 VAC power and to safety related loads and non safety related loads.
2. To provide uninterruptible 120 VAC power to systems that are not safety related.

PURPOSES OF DC POWER SYSTEM

1. To provide highly reliable 125 VDC and 24 VDC power to emergency buses and to equipment required for safe shutdown of the plant.
2. To provide 125 VDC power to balance of plant loads.

ELECTRICAL SYSTEMS

(9.1 - 9.4)

LESSON OBJECTIVES

1. Explain the purpose of the following major system components:
   a. Unit generator
   b. Load break switches
   c. Main transformer
   d. Reserve Station Service Transformer
   e. Normal Station Service Transformer
2. Explain the arrangement of electrical divisions.
3. List typical Emergency Core Cooling System (ECCS) loads that are powered from the 4.16 Kv emergency buses.
4. List and explain the reasons for the diesel generator automatic start signals.
5. Explain the possible sources of power to the 4.16 Kv emergency buses.
6. Explain why load sequencing is necessary under accident conditions.
7. Explain the interfaces Emergency AC Power (EP) System has with the following plant systems:
   a. Normal AC Power (NP) System
   b. DC Power System
   c. 120 VAC Power System
8. Explain why failure of the DC Power System is listed as a major contributor to core damage frequency.

EMERGENCY CORE COOLING SYSTEMS (ECCS)

(10.0)

PURPOSE

To provide core cooling under Loss of Coolant Accident (LOCA) conditions to limit fuel cladding damage and therefore limit the release of radioactive materials to the environment.

LESSON OBJECTIVES

1. State the systems' purpose.
2. List the ECCS acceptance criteria.
3. Explain the electrical division assignments of the ECCS.
4. Describe the integrated ECCS response to small, intermediate, and large break Loss of Coolant Accidents.

HIGH PRESSURE COOLANT INJECTION SYSTEM

(10.1)

PURPOSES

1. To maintain reactor vessel inventory after small breaks which do not depressurize the vessel.
2. To backup the Reactor Core Isolation Cooling (RCIC) System under reactor vessel isolation conditions.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain the major system flow paths.
3. List the automatic initiation signals.
4. Explain the system's response to an automatic initiation signal.
5. State the difference between a HPCI trip and a HPCI isolation.
6. State the basis of the precaution that warns against prolonged operation of the HPCI system below 2200 RPM.
7. State the condition under which the manual isolation push button may be used to produce an isolation.
8. Concerning the HPCI system flow controller, state the parameter that is controlled in the following modes. a. Manual b. Automatic
9. Explain the interfaces this system has with the following systems or components:
   a. Condensate storage tank
   b. Primary Containment System
   c. Reactor Core Isolation Cooling System
   d. Emergency AC Power System
   e. DC Power System
   f. All other ECCS

AUTOMATIC DEPRESSURIZATION SYSTEM

(10.2)

PURPOSE

To depressurize the reactor vessel so that the low pressure Emergency Core Cooling Systems (ECCS) can inject water to mitigate the consequences of a small or intermediate size Loss of Coolant Accident (LOCA) should the High Pressure Coolant Injection System fail.

LESSON OBJECTIVES

1. State the system's purpose.
2. List the automatic initiation signals.
3. List the manual signals that will open the SRVs.
4. Explain the system's response to an automatic initiation.
5. Explain the manual override capability of the system.
6. Explain the interfaces this system has with the following systems:
    a. Main Steam System
    b. Primary Containment System
    c. All other ECCS
    d. DC Power System

CORE SPRAY SYSTEM

(10.3)

PURPOSE

To provide spray cooling to the reactor core to help other Emergency Core Cooling Systems (ECCS) mitigate the consequences of Loss of Coolant Accidents when reactor pressure is low enough for the system to inject water.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain the major system flow paths.
3. Provide the purpose(s) of the major system components.
   a. Pumps
   b. Minimum flow valve
   c. Injection valve
   d. Testable check valve
   e. Spargers
4. List the automatic initiation signals.
5. Explain the system's response to an automatic initiation.
6. Explain the interrelations this system has with the following systems:
   a. Reactor Vessel System
   b. Primary Containment System
   c. All other ECCS
   d. Emergency AC Power System

RESIDUAL HEAT REMOVAL SYSTEM

(10.4)

PURPOSES

1. To restore and maintain desired water level in the reactor vessel following a Loss of Coolant Accident (LOCA).
2. To condense steam and reduce airborne activity in the containment following a LOCA.
3. To remove heat from the suppression pool.
4. To remove decay heat from the core following a reactor shutdown.
5. To condense reactor steam and return the condensate back to the reactor vessel via the Reactor Core Isolation Cooling (RCIC) System.
6. To provide fuel pool cooling if capacity beyond the normal system is required.
7. To flood the containment if required for long term post LOCA recovery operations.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain the purpose of major system components.
   a. Suction valves (suppression pool and shutdown cooling)
   b. RHR pumps
   c. Minimum flow valve
   d. RHR heat exchanger
   e. Containment spray spargers
   f. Line fill
3. Explain the flow path for each mode of operation.
4. List the automatic and manual initiation signals for the Low Presssure Coolant Injection mode of operation.
5. Explain the interfaces this system has with the following systems:
   a. Primary Containment System
   b. Reactor Vessel System
   c. All other ECCS
   d. Emergency AC Power System

CIRCULATING WATER SYSTEM

(11.1)

PURPOSE

To provide cooling water for the station main condensers.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purpose.

REACTOR BUILDING SERVICE WATER SYSTEM

(11.2)

PURPOSE

To transfer heat from the reactor building components to the Long Island Sound, and provide an emergency source of cooling water to the reactor vessel and spent fuel pool.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purpose.
3. Describe the systems response to a Loss of Coolant Accident.

REACTOR BUILDING CLOSED LOOP COOLING WATER SYSTEM

(11.3)

PURPOSES

1. To transfer heat from the components cooled by the RBCLCWS to the Reactor Building Service Water System via heat exchangers.
2. To provide cooling water to reactor auxiliary equipment and other miscellaneous reactor building equipment during normal operation.
3. To provide nuclear safety related systems with a redundant means of cooling during an accident condition in order to accomplish and maintain a safe shutdown.

LESSON OBJECTIVES

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Describe the systems response to a Loss of Coolant Accident.

TURBINE BUILDING SERVICE WATER SYSTEM

(11.4)

PURPOSE

To transfer heat from non-safety related components in the turbine building to the Long Island Sound.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purpose.

TURBINE BUILDING CLOSED LOOP COOLING WATER SYSTEM

(11.5)

PURPOSE

To transfer heat from non-safety related components in the turbine building, radwaste building, and service building to the Turbine Building Service Water System.

LESSON OBJECTIVES

1. State the system's purpose.
2. Explain how the system accomplishes its purpose.

FUEL POOL COOLING AND CLEANUP SYSTEM

(12.1)

PURPOSE

1. Remove decay heat released from the spent fuel elements.
2. Maintain water quality for refueling activities and storage of spent fuel.
3. Provide shielding to reduce radiation levels on the refueling floor.

LESSON OBJECTIVES

1. State the system's purpose and describe how the system accomplishes its purpose.
2. Describe the design features of the system which prevent inadvertently lowering the water level in the spent fuel pool .